GNEP-01-08

TOPICAL REPORT

Advanced Fuel Cycle Facility Accident Analyses as Part of the Global Nuclear Energy Partnership Programmatic Environmental Impact Statement

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May 2008
Revision 0

TETRA TECH, INC.

GNEP-01-08
Rev. 0

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May 2008

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Abbreviations and Acronyms

AFCF	Advanced Fuel Cycle Facility
ALOHA	Areal Location of Hazardous Atmospheres
ARF	airborne release fraction
BG	block-group
DOE	Department of Energy
DR	damage ratio
EA	environmental assessment
EIS	environmental impact statement
EMT	electrometallurgical treatment
ERPG	Emergency Response Planning Guidelines
F&ORs	functional and operational requirements
FMEF	Fuels and Materials Examination Facility
FPR	Fuel Processing Restoration
FR	fast reactor
GNEP	Global Nuclear Energy Partnership
HEPA	high efficiency particulate air
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
LANL	Los Alamos National Laboratory
LPF	leak path factor
LWR	light water reactor
MACCS2	MELCOR Accident Consequence Code System 2
MAR	material-at-risk
MEI	maximally exposed individual
MOX	mixed oxide
NE	Nuclear Energy
NEPA	National Environmental Policy Act
NIW	noninvolved worker
NPR	New Production Reactor
ORR	Oak Ridge Reservation
ORPS	Occurrence Reporting & Processing System
PEIS	programmatic environmental impact statement
RF	respirable fraction
SIH	standard industrial hazard
SNF	spent nuclear fuel
SRS	Savannah River Site
TBP	tri-n-butyl phosphate

TRU	transuranic
USCB	US Census Bureau

TOPICAL REPORT

Advanced Fuel Cycle Facility Accident Analysis as Part of the Global Nuclear Energy Partnership Programmatic Environmental Impact Statement

1. INTRODUCTION

The Department of Energy's (DOE) Office of Nuclear Energy (NE) is responsible for implementing the Global Nuclear Energy Partnership (GNEP), which seeks to develop worldwide consensus on enabling expanded use of economical, carbon-free nuclear energy to meet growing electricity demand. GNEP will use a nuclear fuel cycle that enhances energy security, while promoting non-proliferation. It would achieve its goal by having nations with secure, advanced nuclear capabilities provide fuel services — fresh fuel and recovery of used fuel — to other nations who agree to employ nuclear energy for power generation purposes only. The closed fuel cycle model envisioned by this partnership requires development and deployment of technologies that enable recycling and consumption of long-lived radioactive waste. The Advanced Fuel Cycle Facility (AFCF) would be used to perform the research and development activities necessary to demonstrate the critical technologies needed to change the way spent nuclear fuel (SNF) is managed – to demonstrate recycling technologies that enhance energy security in a safe and environmentally responsible manner, while simultaneously promoting non-proliferation.

Pursuant to the *National Environmental Policy Act* (NEPA) of 1969, as amended (42 USC 4321 et seq.), and the DOE Regulations Implementing NEPA (10 CFR Part 1021), NE is preparing a Programmatic Environmental Impact Statement (PEIS) that includes the AFCF in order to decide: (1) whether to proceed with the AFCF; and (2) if so, where to locate the AFCF. Documents prepared under NEPA should inform the decision maker and the public about the chances that reasonably foreseeable[1] accidents associated with proposed actions and alternatives could occur, and about their potential adverse consequences. Accident analyses are necessary for a reasoned choice among the proposed action and alternatives and appropriate consideration of mitigation measures. Accident analyses in NEPA documents can provide estimates of the magnitude of risk[2] that the proposed action and alternatives would present and a comparison of risk among the proposed action and alternatives.

This report describes how locations or operations were selected for analysis, the computer codes used to estimate consequences, the development of the scenarios and assumptions about source terms, the selection of computer modeling, a description of the results, and predicted health effects for the AFCF in the GNEP PEIS. This methodology follows the general guidance provided by DOE in *Recommendations for Analyzing Accidents under the National Environmental Policy Act* (DOE 2002d). The report also provides the results of these analyses.

Section 2 presents background information related to the AFCF and its activities. Section 3 presents general NEPA guidance used in this analysis. Section 4 presents an overview of the methodology used to identify candidate scenarios. Section 5 describes the scenarios selected for analysis and describes a

[1] The term "reasonably foreseeable" not only includes events that may be expected, but extends to events that may have catastrophic consequences, even if their probability of occurrence is low, provided that the analysis of the impacts is supported by credible scientific evidence, is not based on pure conjecture, and is within the rule of reason. (Council on Environmental Quality NEPA Regulations, 40 CFR 1502.22)

[2] Risk, as used here, refers to the combination of the probability and consequences of an accident. When risk cannot be quantified, it is appropriate to discuss risk qualitatively in terms of the probability and potential consequences. (DOE 2002d)

release estimate. Section 6 presents the methodology used for consequence calculation for the selected accidents. Section 7 provides the consequence analysis results.

2. BACKGROUND

The AFCF will be used to develop and demonstrate the nuclear fuel cycle technologies necessary to meet the requirements of a sustainable United States (U.S.) nuclear industry for the next 50 years (WGI 2008a). The AFCF will have the capabilities necessary to develop and test fuel cycle processes and individual equipment items for: 1) separating the components of spent nuclear fuels (SNF), including recycled fuels and 2) fabricating advanced proliferation resistant, recycle fuel types. The facility will also support the resolution of design issues related to process control and integration, reliability and scale-up via integrated systems, operational tests at bench-scale, and at engineering scale.

The core of the AFCF will include 1) large shielded and remotely maintained areas to validate/demonstrate spent fuel treatment, fuel fabrication processes, and safeguards monitoring, and 2) small shielded and remotely maintained areas (e.g., hot cells and gloveboxes – automated and/or manual) to conduct development activities and validate bench-scale unit operations, with all systems fully integrated and operating at engineering-scale. This development process will provide critical path information for the design of future full-scale production facilities and the confidence to know that production scale processes will perform as intended.

The primary functions of the AFCF are listed below and described in the *High-Level Functional & Operational Requirements for the Advanced Fuel Cycle Facility* (referred to hereafter as the AFCF F&ORs) (Ridgway 2006).

- Materials receipt and storage
- SNF separations (including preparation and head-end treatment)
- Separations material handling and storage
- Separations material conditioning
- Fuel fabrication
- Scrap recovery
- Waste handling, treatment, and storage
- Shipping
- Material Control & Accountancy
- Facility support

Because of the 50-year time horizon of the AFCF and its mission of researching, developing, and demonstrating new nuclear technologies, it is not possible to identify all of the technologies that may be used in the 50-year life of the AFCF for each of these functions. Similarly, the AFCF will support research, development and demonstration of the separations and/or fabrication of the following fuel types:

- Mixed uranium and transuranic (TRU) ceramic fuel for use in fast reactors (FRs)
- Mixed uranium and TRU metal alloy fuel for use in sodium-cooled fast reactors
- Mixed nitride fuel for use in FRs
- Coated particle fuel for use in very high temperature gas-cooled reactors
- Gas-Cooled Fast Reactor fuel
- Inert matrix fuel for use in light water reactors (LWRs) or FRs
- Sphere-pac or vibropack targets for use in LWRs or FRs
- Uranium-oxide fuel and mixed oxide (MOX) fuel for use in LWRs

Two separations processes, aqueous separations and electrochemical separations, are expected to serve as the initial core of AFCF. The electrochemical process is also referred to as electrometalurgical treatment or electroprocessing. Details of both of these processes, especially the aqueous process, are likely to be revised and modified over the life of the facility. Therefore, flowsheets are presented in simplified form in Figures 2-1 and 2-2. Multiple different fuel fabrication processes are expected to be utilized over the life of the AFCF. Figure 2-3 presents a simplified representation of the fuel fabrication flow sheet. The specific processes involved in feed preparation and fuel element fabrication differ depending upon the type of fuel fabricated.

AFCF includes treatment of all ventilation systems discharges that have the potential to be contaminated with radioactive material. Each process vessel, process cell, and operating area that has the potential to be contaminated includes ventilation treatment. These ventilation treatment systems include condensers, evaporators, pre-filters, and high efficiency air particulate (HEPA) filters as appropriate. The outlet for all process vessels and process cells undergo at least two separate banks of HEPA filtration in series before being discharged to the stack.

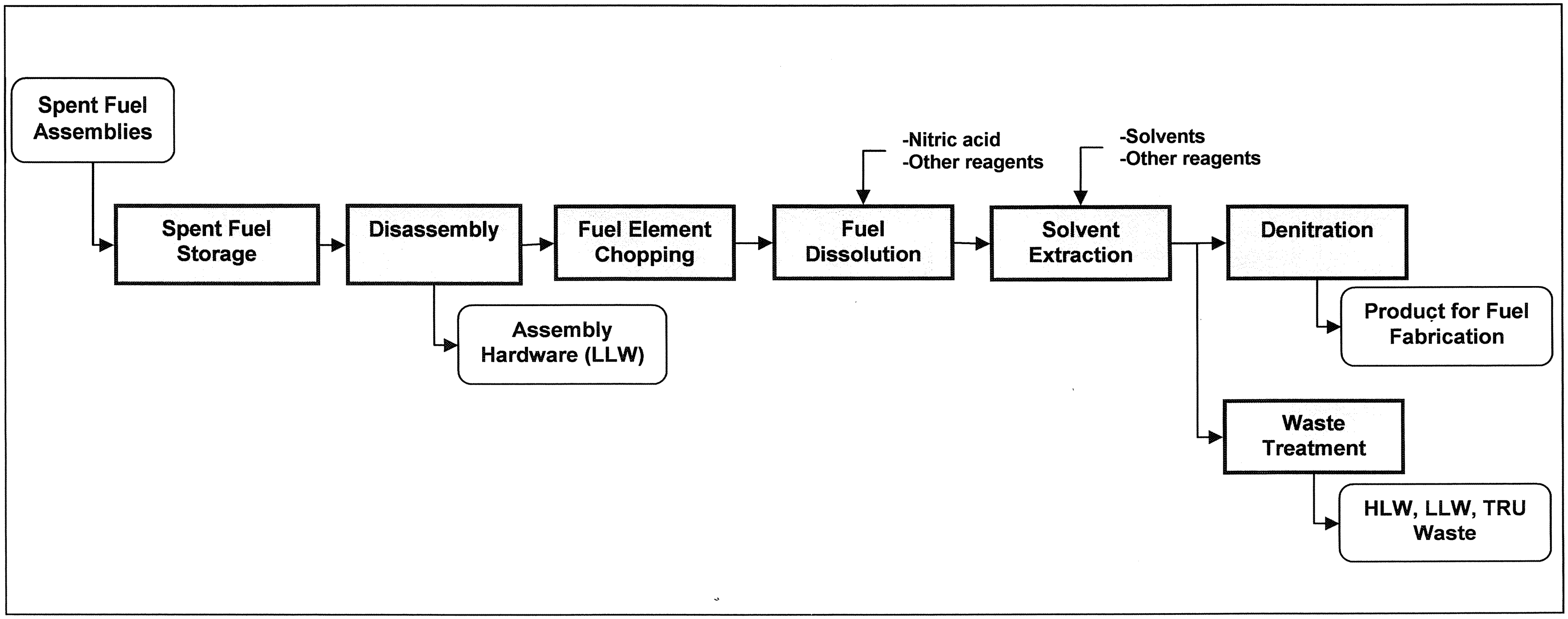

FIGURE 2-1-*Representative Aqueous Process Flow Sheet.*

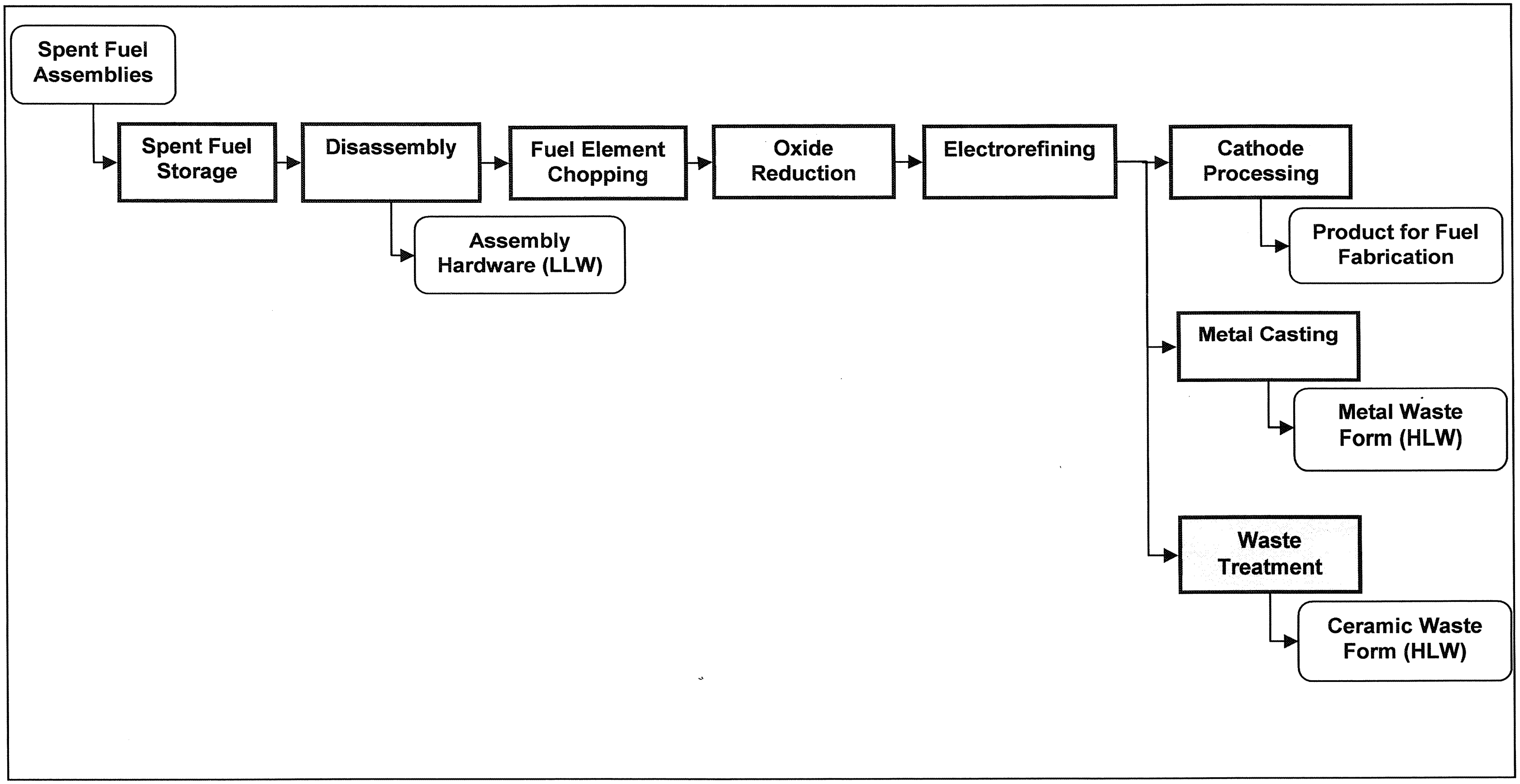

FIGURE 2-2-*Representative Electrochemical Process Flow Sheet*

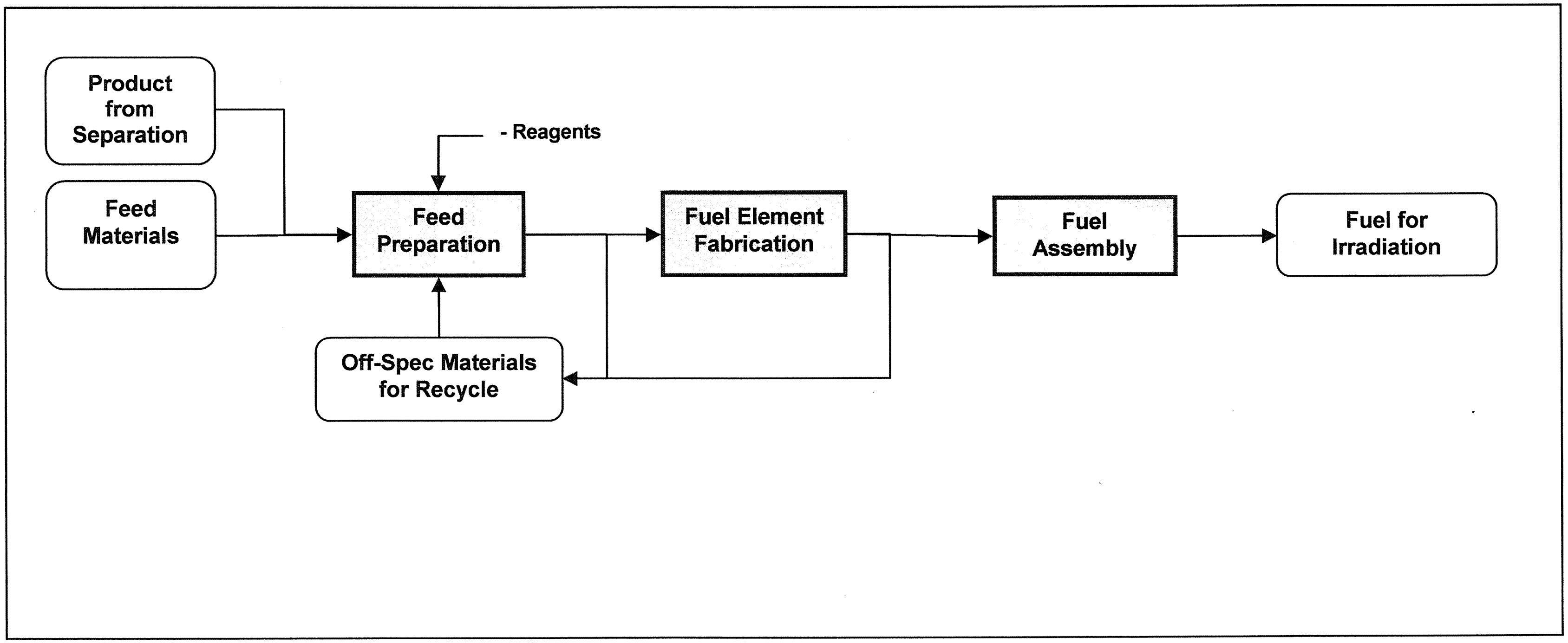

FIGURE 2-3-*Representative Fuel Fabrication Process Flow Sheet*

3. GENERAL GUIDANCE

DOE has issued *Recommendations for Analyzing Accidents Under the National Environmental Policy Act* (DOE 2002d), which provides general guidance on the accident analyses for NEPA documents. In addition to the DOE NEPA accident analysis guidance, the methodology used for analysis of AFCF facility accidents relies heavily upon the DOE guidance for the authorization basis of nonreactor nuclear facilities. The authorization basis guidance considered includes the following:

- *DOE Handbook - Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*, DOE-HDBK-3010-94, Change Notice #1 (DOE 2000i)
- *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Documented Safety Analyses*, DOE-STD-3009-94, (Change Notice 2, April 2002), (DOE 2006a)
- *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*, DOE-STD-1027-92, (Change Notice 1, September 1997) (DOE 1997)

However, the purpose of accident analysis for NEPA differs from the purpose for facility authorization bases in several important respects. One purpose of the accident analysis for the authorization basis is to provide reasonable assurance that a DOE nuclear facility can be operated safely by defining and controlling commitments for design, procurement, construction, and operation. To accomplish that purpose, the DSA and accompanying hazard controls require substantially greater details of design and specific operations than are usually available when NEPA documents are prepared. Accident analyses in NEPA documents inform the decision maker and the public of the nature of the risk associated with the proposed action and alternatives. Therefore, NEPA accident analyses typically focus on a limited number of bounding consequence scenarios that cover the spectrum of likelihoods and include mitigative features. Also, NEPA documents are frequently prepared early in the life cycle of proposed facilities, when only conceptual design information is available, and usually precede authorization basis documents. As a result, accident analyses for NEPA documents tend to address more generalized scenarios.

DOE's accident analysis guidance (DOE 2002d) does not provide a "cookbook" approach, but rather advises document preparers to use a sliding scale approach to accident analyses. This sliding scale approach includes consideration of the appropriate range and number of accident scenarios to consider, and the level of analytical detail and degree of conservatism that should be applied. The following paragraphs address the general approaches for analytical detail and conservatism.

Analytical detail – The AFCF is an R&D facility intended to explore and demonstrate new technologies over a 50 year life. It is not possible to foresee all technologies and operations that may be undertaken in support of this mission. Consequently, it is more appropriate to ensure that the analysis addresses the full breadth of activities that may be undertaken rather than focusing on details of select scenarios. Therefore, this analysis addresses the spectrum of accident types foreseeable at AFCF without necessarily focusing on the specific initiating events or details of the scenario progression.

Conservatism – Bounding approaches based on conservative assumptions have several advantages over more realistic analysis, including: streamlining the analysis when there are many uncertainties, avoiding the need to prepare more realistic analyses when they are not warranted, and being more defensible because they are unlikely to underestimate potential accident consequences. Bounding approaches will be used for AFCF as long as they do not mask differences among alternatives, provide less information about the potential need for mitigation, or result in a misleading presentation of accident risks. Bounding approaches will not affect technology decisions for the AFCF because there are no competing technology alternatives. Since the bounding approaches used for the AFCF accident analysis are reasonable and applied to all sites, they do not invalidate site selection. The degree of conservatism will be discussed qualitatively in order to provide insight into the expected risks.

4. METHODOLOGY FOR SCENARIO SELECTION

The goal of the accident analyses is to develop realistic accident scenarios that address a reasonable range of event probabilities and consequences in order to inform the decision maker and the public of the AFCF accident risks. The maximum reasonably foreseeable accidents are analyzed to represent potential accidents at the high consequence end of the spectrum. A maximum reasonably foreseeable accident is an accident with the most severe consequences that can reasonably be expected to occur for a given proposal. The steps involved in selecting and defining scenarios to be analyzed are as follows:

1) Assemble and review available information and technical resources applicable to the AFCF buildings, equipment, processes and operations.

2) Identify potential hazardous conditions and define a preliminary set of candidate accidents.

3) Select a final set of accidents, develop scenarios, and derive applicable data for the AFCF accident analysis in the GNEP PEIS including frequency and release parameters (source term, release duration, and release point).

Each of these steps is addressed in greater detail in the following subsections.

4.1 Available Information

The first step in the accident analysis process is the assembly and review of available information. NEPA documents frequently rely heavily on authorization basis documents as the basis for the identification of candidate accidents. Authorization basis documents are not available for AFCF so this accident analysis also relies on information from related activities to ensure a comprehensive identification of accidents.

The following information sources are used in the identification of candidate accidents.

1. *AFCF design and operations information* – AFCF information is reviewed to define the scope and nature of activities, and identify material inventories and potential hazards. The following AFCF documents were used in the accident analysis:
   - *High-Level Functional & Operational Requirements for the Advanced Fuel Cycle Facility,* (Ridgway 2006)
   - *Advanced Fuel Cycle Facility Conceptual Design and NEPA Support Activities*, (WGI 2008a), hereafter referred to as the AFCF NEPA Data Study
   - *Evaluation of the Distributed Facility Option for the Advanced Fuel Cycle Facility*, (WGI 2008b), hereafter referred to as the AFCF Distributed Facility Study
   - *Draft 30% Conceptual Design Report For the Advanced Fuel Cycle Facility,* (DOE 2007), especially Section 8 – Safety and Appendix E – Preliminary Hazards Analysis.

2. *Relevant NEPA documents* – DOE has a long history of nuclear fuel cycle activities and there are numerous DOE NEPA documents for activities similar to those of the proposed AFCF. For example, there are several NEPA documents that address SNF management both on a programmatic and a site level. These NEPA documents identify the full range of accidents considered appropriate for the scope of their activities. In general, the scope of each of these related activities covers only a portion of the AFCF activities; however, collectively they cover the full scope of AFCF activities. The AFCF function covered by each related NEPA document is identified. The NEPA documents for related activities are reviewed to identify their bounding accidents that are relevant to AFCF. A list of these relevant accidents is then developed.

3. *Occurrence Reporting & Processing System (ORPS) database* – The ORPS database (DOE 2007u) is reviewed in order to gain insights into types of accidents that have occurred at facilities with similar operations and to ensure that all of these accidents are addressed. In most cases, the ORPS events do not result in worker or public consequences and do not warrant evaluation. However, the ORPS evaluation ensures consideration of scenarios that have occurred at similar activities but may not have been covered by the related NEPA documents. Any new scenarios identified in this process are added to the list of candidate scenarios.

4. *Hazard checklist* –Various hazards checklists have been developed to support development of authorization basis documents. There are similarities among these hazards checklists and they differ primarily in the level of detail presented. A standard hazard checklist (see Appendix A) is reviewed to ensure that all hazard types have been considered. This review is primarily performed to ensure that non-radiological hazards are adequately considered. Any new scenarios identified in this process are added to the list of candidate scenarios.

A number of NEPA documents address functions similar to those of AFCF. The following NEPA documents are considered especially relevant to the AFCF activities and are used as the basis for identifying candidate scenarios:

- *Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel*, DOE/EIS-0306F, August 2000, (DOE 2000e), referred to hereafter as the EMT EIS

- *Environmental Assessment: Fuel Processing Restoration at the Idaho National Engineering Laboratory*, DOE/EA-0306, August 1987, (DOE 1987), referred to hereafter as the FPR EA

- *Idaho High-Level Waste & Facilities Disposition Environmental Impact Statement*, DOE/EIS-0287F; September 2002, (DOE 2002e), referred to hereafter as the IHLW EIS

- *Environmental Impact Statement on the Construction and Operation of a Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina*, NUREG-1767, January 2005, (NRC 2005c), referred to hereafter as the MOX EIS

- *Environmental and Other Evaluations of Alternatives for Siting, Construction, and Operating New Production Reactor Capacity*, DOE/NP-0014, September 1992, (DOE 1992c), referred to hereafter as the NPR Rpt.

- *Accident Assessments for Idaho National Engineering Laboratory Facilities*, DOE/ID-10471, March 1995, (DOE 1995a), referred to hereafter as the PSNF EIS

- *Savannah River Site Spent Nuclear Fuel Management Environmental Impact Statement*, DOE/EIS-0279, March 2000, (DOE 2000f), referred to hereafter as the SRS SNF EIS

Table 4.1-1 identifies each AFCF function as identified in Section 3 of the *AFCF F&ORs* (Ridgway 2006) and identifies the NEPA documents that address similar functions. The SNF separations function has been subdivided below because some of the NEPA documents only cover a portion of the AFCF separations scope and it is important to ensure complete coverage. As shown by Table 4.1-1, multiple NEPA documents were reviewed for each AFCF function.

TABLE 4.1-1-*AFCF Functions Addressed by Other NEPA Documents*

Function	EMT EIS (DOE 2000e)	FPR EA (DOE 1987)	IHLW EIS (DOE 2002e)	MOX EIS (NRC 2005c)	NPR Rpt. (DOE 1992)	PSNF AA (DOE 1995a)	SRS SNF EIS (DOE 2000f)
Materials receipt and storage	✓	✓		✓	✓	✓	✓
Spent Fuel Separations							
• Aqueous head-end	✓				✓		✓
• Aqueous separations	✓	✓			✓		✓
• Electrochemical separations	✓					✓	✓
Separations material handling and storage	✓	✓			✓		✓
Separations material conditioning	✓	✓			✓		✓
Fuel fabrication				✓	✓		
Scrap recovery				✓	✓		
Waste handling, treatment, and storage	✓	✓	✓		✓	✓	✓
Shipping	✓	✓	✓	✓	✓	✓	✓
Material Control & Accountancy	✓	✓		✓	✓	✓	✓
Facility support	✓	✓		✓	✓	✓	✓

4.2 Identification of Candidate Accidents

After review of the available information, a list of candidate accidents to be considered for analysis is developed and presented in Appendix B. The list is not intended to be a comprehensive list of all specific accidents that could occur at AFCF, as might be the case for an authorization basis accident analysis; but rather, it is a list of candidates that bounded other AFCF accidents. Each relevant NEPA document has already selected the bounding scenarios for its scope of activities, so it is not necessary to reconsider all potential accidents. For example, if a facility-wide fire has been listed, it is not necessary to list less-impacting fires that are expected to have comparable or lower consequences. The scope of the accidents to be considered, the types of initiators, and accident phenomena types are discussed in the following subsections.

Table 4.1-1 shows that all AFCF functions are addressed by one or more other NEPA documents. While all functions are covered, a more detailed review was performed to determine if there are process differences that might warrant further evaluation. The following paragraph address the process differences that might affect the selection of accidents.

- *Voloxidation* – AFCF would include a voloxidation step not explicitly included in the other NEPA documents. The voloxidation step converts the UO_2 pellets to a U_3O_8 powder that is considerably more dispersible. Fires and explosions are already considered for the head-end, so voloxidation does not result in a new accident type, but it may affect the consequences. The evaluation of consequences will take into account the potential dispersibility of the voloxidation product.

- *Partial separations* – AFCF aqueous processing would include multiple partial separations steps not specifically included in the other NEPA documents. These partial separations processes include UREX for uranium and technetium extraction, CCD-PEG for cesium and strontium extraction, TRUEX for transuranic and lanthanide extraction, and TALSPEAK for partitioning of fission products from transuranics. No new accidents have been identified as a result of these process differences, though the composition of the material at risk will be affected. Since all separations steps after the initial step involve a subset of the original inventory, it will be conservative to base analyses on the full SNF inventory prior to separations.

- *Equipment differences* – The AFCF aqueous separations process is expected to use centrifugal contactors rather than extraction columns, which were the basis in at least some of the other NEPA documents. Centrifuges are smaller and contain a smaller volume of fuel than the extraction columns, so the consequences of a given accident may be lower. This analysis is conservatively based on the overall volume of dissolved fuel in the extraction system in order to cover either equipment option and is not necessarily based on the volume in a contactor.

The AFCF preliminary hazards analysis presented in Appendix E of the *Draft 30% Conceptual Design Report For the Advanced Fuel Cycle Facility* (DOE 2007) was reviewed to deterimine if it identified accidents whose consequences exceed the bounding accidents identified in the related NEPA documents. The preliminary hazard analysis was performed to support the authorization basis and reports unmitigated[3] consequences, so its results cannot be compared directly with the mitigated consequences reported in other NEPA documents. After accounting for the differences in methodology, it was concluded that the preliminary hazards analysis does not identify scenarios that would bound those selected in other NEPA documents.

4.2.1 Scope of Accidents Considered

The analysis considers both radiological and non-radiological accidents. Radiological accidents include the release of radioactive material or exposure of workers to high radiation fields. Non-radiological accidents include release of chemically hazardous materials and hazards, as well as unusual hazards (e.g., high-energy lasers and high explosives) that are not standard industrial hazards. Standard industrial hazards (SIHs) are adequately addressed by DOE-prescribed programs and DOE or national consensus codes or standards. Standard industrial hazards are considered for their potential as initiators but are dismissed from further consideration for their direct impacts.

The accident analysis considers accident scenarios that represent the spectrum of reasonably foreseeable accidents, including low probability/high consequence accidents and higher probability/(usually) lower consequence accidents. Typically, accidents with a frequency of less than 10^{-7} per year are not reasonably foreseeable and rarely need to be examined.

Intentional destructive acts are acts of sabotage or terrorism whose physical effects can include fire, explosion, missile or other impact force. The impacts of an intentional act may be similar to the effects of non-intentional accidents. Analysis of such intentional acts is beyond the scope of this analysis and may be performed in a separate study.

[3] Unmitigated means considering the material quantity, form, location, dispersibility and interaction with available energy sources, but not considering safety features (e.g., ventilation system, fire suppression, etc.) which will prevent or mitigate a release.

4.2.2 Accident Initiators Types

An accident is a sequence of one or more unplanned events with potential outcomes that endanger the health and safety of workers and the public. An accident can involve a combined release of energy and hazardous materials (radiological or chemical) that might cause prompt or latent health effects. The sequence usually begins with an initiating event, such as a human error, or equipment failure, or an earthquake, potentially in combination with other events and conditions that could be dependent or independent of the initial event. The combination of the initiating event and contributing events and conditions dictate the accident's progression and the extent of materials released. Initiating events fall into three categories:

- *Internal initiators* normally originate in and around the facility, but are always a result of facility operations. Examples include equipment or structural failures and human errors.

- *External initiators* are independent of facility operations and normally originate from outside the facility. Some external initiators affect the ability of the facility to maintain its confinement of hazardous materials because of potential structural damage. Examples include aircraft crashes, vehicle crashes, nearby explosions, and toxic chemical releases at nearby facilities that affect worker performance.

- *Natural phenomena initiators* are natural occurrences that may affect the facility and its operations. Examples of natural phenomena hazards (NPH) include earthquakes, high winds, floods, lightning, and snow. Natural phenomena initiators can also affect nearby facilities, which in turn may affect the primary facility under review. Earthquake typically bounds other natural phenomena. Earthquakes generate severe lateral and vertical stresses upon the structure and equipment that may result in confinement failure, breach, or collapse. Seismic forces may cause material spills but do not generate gas flow to transport particulate materials, although flows are generated by falling debris or any fires/explosion caused by the seismic event.

The AFCF accident analysis explicitly considers internal, external, and natural phenomena initiators which may result in any of a variety of accident types. Intentional destructive acts are only considered in a general comparative manner.

4.2.3 Accident Phenomena Types

Accidents can also be characterized by the type of accident phenomena that produces the release of radioactive or chemically hazardous materials. The accident types have different potentials for facility damage, release, and dispersion. AFCF accidents will be grouped into the following accident types consistent with DOE-HDBK-3010-94 (DOE 2000i). Except for criticality, each of these accident types may apply to both radioactive and chemically hazardous materials.

- *Criticality* events produce high radiation fields and generate fission products that may become airborne. Fission product gases are released from liquid criticalities and from solid criticalities to the extent the underlying critical mass is degraded. Solid fission products typically have small release fractions determined by the degree of physical stress placed on the critical mass itself. At large fission yields, solid critical masses may experience some degree of melting or oxidation. In addition to the generated fission products, a nuclear criticality may also release pre-existing radioactive material.

- *Explosions* may result from chemical (e.g., oxidations involving branch-chain products, oxidations of gas-oxidant mixtures) or physical (overpressurization of tanks or vessel, vapor explosions) reactions.

Explosions generate shock and blast effects with potential for gas flow subsequent to the explosive event that may subdivide and entrain material. Shock waves are supersonic pressure waves (pulses) that can transmit an impulse to materials and the surrounding structures resulting in shattering of solid items, but do not result in significant dispersion.
Blast effects are typically subsonic and involve material entrained in the gas flow. Blast effects are often more damaging. The gas expanding from the explosion zone carries material from the explosion site. If the explosion is adjacent to the material, then blast effects can cause damage above and beyond the initial impulse loading. Some explosive reactions may be followed by chemical reactions, material vaporization, or fires that lead to substantial gas flows following the explosive event. These gas flows may also entrain material. Deflagrations do not involve shock, but can simulate blast effects. Under proper conditions (e.g., confinement, structural features that enhance turbulence), deflagrations can transition to detonations and produce shock waves.

- *Fire* generates heat and combustion gases that may affect the radioactive material and/or the materials upon which they are deposited, compromise barriers, and/or pressurize containers/enclosure that may lead to the airborne release of contained radioactive materials. Released material is then entrained in general convective currents that provide transport for particulate materials.

- *Spills* typically involve failure of a containment/confinement barrier as of a drop or toppling event. In addition to the loss of containment/confinement, drops from substantial heights may also result in sub-division of the material due to the impact. Airflow from the event may result in limited suspension and transport the particles.

- *Earthquake* typically bounds other natural phenomena in terms of consequences. Earthquakes generate severe lateral and vertical stresses upon the structure and equipment that may result in confinement failure, breach, or collapse. The response of the materials-of construction may dislodge materials-of-concern by vibration, impact of debris, and fragmentation. Seismic forces may cause material spills but do not generate gas flow to transport particulate materials, although flows are generated by falling debris or any fires/explosion caused by the seismic event. Earthquakes are actually accident imitators, but they are also identified here as a separate accident type because of the extreme energies involved and unique potential combinations of resulting phenomena that may result.

4.2.4 Source Term

The source term is the amount of material, in grams or curies, released to the air. This section summarizes the methodology described in Section 1.2 of DOE-HDBK-3010-94 (DOE 2000i) for calculation of the source term. The source term is calculated by the equation:

Source Term = MAR x ARF x RF x DR x LPF, where:

MAR — Material-at-Risk: the amount of radioactive materials (in grams or curies of activity for each radionuclide) available to be acted on by a given physical stress.

DR — Damage Ratio: the fraction of MAR impacted by the actual accident-generated conditions under evaluation.

ARF — Airborne Release Fraction: the coefficient used to estimate the amount of a radioactive material that can be suspended in air and made available for airborne transport under a specific set of induced physical stresses.

RF — Respirable Fraction: the fraction of airborne radionuclides as particles that can be transported through air and inhaled into the human respiratory system and is commonly assumed to include particles 10-μm Aerodynamic Equivalent Diameter (AED) and less.

LPF — Leak Path Factor: the fraction of airborne materials transported from containment or confinement deposition or filtration mechanism (e.g., fraction of airborne material in a glovebox leaving the glovebox under static conditions, fraction of material passing through a HEPA filter).

Values used in the AFCF accident analysis are based on guidance provided in the appropriate sections of DOE-HDBK-3010-94 for the type of materials and event phenomena involved. DOE-HDBK-3010-94 frequently provides median and bounding values for these parameters. The bounding values are generally used for the AFCF analyses.

Nuclear criticality events generate fission products that need to be taken into account. In order to calculate the fission products generated, it is necessary to know the type of material involved (e.g., low-enriched uranium) and to estimate the number and timing of fissions that occur. The methodology presented in Section 6.2.3 of DOE-HDBK-3010-94 is used to estimate the initial and total number of fissions occurring for each criticality. The radionuclides generated by the criticality event are based on NRC Regulatory Guides, as presented in Section 6.3 of DOE-HDBK-3010-94.

4.3 Selection of Accidents for Analysis

From the list of candidate accidents (see Appendix B), a set of bounding accidents is identified for analysis. The selection process includes a qualitative assessment of the frequency and consequences of each candidate. Based on the frequency and consequence estimates, most of the candidate accidents are screened from further consideration. The accident selection process involves a combination of data, evaluation, and engineering judgment. The basis for dismissing each candidate is briefly documented.

Four general guidelines, listed below, are followed in the selection of the AFCF accident scenarios.

1) Potential hazardous and accident conditions should include the largest source terms at risk and conditions for worker and public impacts.

2) The accident scenarios selected should cover a spectrum of accident situations ranging from high probability/low consequence events to low probability/high consequence events.

3) For each probability range, the accident with bounding consequences should be selected as representative for the range.

4) The accident scenarios should reflect differences resulting from site specific initiators, meteorology, characteristics (e.g., distance from site boundary and other adjacent facilities), and use existing facilities.

The following subsections address the assignment of frequency categories and consequences to the candidate accidents.

4.3.1 Accident Frequencies

The AFCF facility designs and operations are not finalized at this time, so, it is not possible to provide precise frequency estimates for each accident scenario. Therefore, a qualitative estimate of the frequency of each AFCF scenario, including both the initiating event and conditional events/conditions, are estimated via frequency ranges. In this analysis four frequency categories are defined. Table 4.3.1-1 presents the ranges of frequencies, return periods, and probability of occurrence during the facility life for each category and is based on Table 3-4 of DOE-STD-3009-94 (DOE 2006p). Here, the frequency estimate includes both the initiating event and conditional events/conditions. The accident analysis considers accident scenarios that represent the spectrum of reasonably foreseeable accidents; including low probability/high consequence accidents and higher probability/ (usually) lower consequence accidents. Typically, accidents with a frequency of less than 10^{-7} per year are not reasonably foreseeable and do not need to be examined.

The AFCF frequency estimates are based on NEPA documents for facilities with similar operations which include consideration of historic operating experience in similar heavily shield facilities. The fidelity of accident frequency estimates are lower when exact facility designs and operations have not been finalized. Also, the number of processes and equipment trains can affect the frequency for some scenarios. Therefore, quantitative frequency estimates are not always available. When only a frequency category is available for an accident scenario, the logarithmic midpoint of the category is used for the risk calculations (i.e., 0.03, 10^{-3}, and 10^{-5} per year are used for the Anticipated, Unlikely, and Extremely Unlikely). A frequency estimate is required for all Beyond Extremely Unlikely scenarios.

TABLE 4.3.1-1. Accident Frequency Categories.

Frequency Category	Frequency Range (/yr)	Return Period (yrs)	Probability During Facility Life (50 yrs)
Anticipated (A)	$10^{-2} \leq f \leq 10^{-1}$	$10^{2} \geq T \geq 10^{1}$	$0.4 \leq P < 1$
Unlikely (U)	$10^{-4} \leq f < 10^{-2}$	$10^{4} \geq T > 10^{2}$	$5x10^{-3} \leq P < 0.4$
Extremely Unlikely (EU)	$10^{-6} \leq f < 10^{-4}$	$10^{6} \geq T > 10^{4}$	$5x10^{-5} \leq P < 5x10^{-3}$
Beyond Extremely Unlikely (BEU)	$f < 10^{-6}$	$T > 10^{6}$	$P < 5x10^{-5}$

4.3.2 Qualitative Accident Consequences

A qualitative estimate of consequences for each candidate accident is provided for use in the screening and selection process based on the release and consequences calculation results reported in the related NEPA documents. While the numeric values reported in the related NEPA documents are reported here, these calculations are not directly related to AFCF. The accident and phenomena are appropriate for AFCF, but the numeric values may be different for AFCF because of facility layout, capacity, and site differences. Therefore, this information is used as input to engineering judgment in the selection of accidents and not as a direct basis for selection.

4.3.3 Selection Process

The selection process involves selection of candidate accidents that represent the largest consequence events over the range of frequencies. The selection process applies the following steps to the list of candidate accidents.

1. For each candidate scenario, identify the appropriate initiating event category: As explained in Section 4.2.2, the initiating event categories are: external (E), internal (I), or natural phenomena (N).

2. For each candidate scenario, identify the accident phenomena type. As explained in Section 4.2.3, the accident phenomena types considered are: criticality (C), explosion (X), fire (F), earthquake (E), and spill (S).

3. For each candidate scenario, identify the appropriate frequency category. When a scenario may overlap two categories, the scenario is placed into the higher frequency category. As explained in Section 4.3.1, the frequency categories are: anticipated (A), unlikely (U), extremely unlikely (EU), and beyond extremely unlikely (BEU).

4. Assign each candidate scenario to one or more relevant operations. The AFCF operations are categorized as aqueous separations processing, electrochemical processing, fuel fabrication, and balance of plant.

5. For each AFCF operation, identify one or more scenarios in each frequency category for each AFCF operation whose consequences represent or bounds the other scenarios in that frequency category. The determination of consequences is a qualitative evaluation as discussed in Section 4.3.2.

6. The list of selected scenarios is then reviewed and adjusted to eliminate repetition of similar scenarios in multiple operations, eliminate scenarios for one operation that bounded by another operation, and ensure that scenarios illuminating alternative differences are retained.

This selection process is iterative and accidents are added, deleted, and modified as additional information and insights become available.

4.3.4 Facility Alternatives Considered

DOE is considering a variety of alternatives for AFCF that are discussed below.

Greenfield Alternative-This alternative involves use of only new facilities at one site and is discussed in the AFCF NEPA Data Study (WGI 2008a). The Greenfield Alternative is the basis for the analyses performed here.

Brownfield Alternative-This alternative involves use of existing and new facilities at one site, as discussed in Appendix A-1 of the AFCF NEPA Data Study (WGI 2008a). The existing facility options being considered are the Fuels and Materials Examination Facility (FMEF) at the Hanford Site and the Idaho Nuclear Technology and Engineering Center (INTEC) at the Idaho National Laboratory (INL). For either of these existing facilities, new process equipment would be required and this new process equipment would be similar to the equipment used in the Greenfield Alternative. Because the processes and equipment are similar, the internally initiated accidents for this alternative are similar to the accidents analyzed for the Greenfield Alternative. The external events accident (i.e., the Aircraft Crash) impacts for a heavily shielded facility in the Brownfield Alternative would be the same as the impacts for a similar heavily shielded facility in the Greenfield Alternative.

The existing facilities may not meet current codes and standards for nuclear facilities (WGI 2008a), but these facilities were constructed to criteria of the 1970's through 1990's, which were not dramatically different from current criteria. Prior to operation, existing facilities would be required to meet DOE safety requirements, so there will not be large differences in risk. The natural phenomena accident (i.e., the Beyond Design Basis Earthquake) may have a slightly higher frequency for this alternative than for the Greenfield Alternative. The Beyond Design Basis Earthquake is not the bounding consequence or risk AFCF event (see Section 3.3), so the overall consequence and risk for this alternative will be approximately the same as the consequence and risk for the Greenfield Alternative. Therefore, the analyses presented for the Greenfield Alternative are used as the basis for the Brownfield Alternative.

Distributed Greenfield Alternative-DOE is also considering distributing the AFCF function across multiple sites with new facilities as addressed in the AFCF Distributed Facility Study (WGI 2008b). The modules considered for distribution are aqueous processing, electrochemical separations process, fuel fabrication, and process support and development. Collectively, these modules would perform the same functions being performed by a centralized AFCF. Each of these modules would have its own waste management capabilities and support capabilities. Distribution of the AFCF functions does not result in new accident types, initiators, or consequences. The analysis performed for the Greenfield Alternative identifies the bounding AFCF scenario for each frequency category for all AFCF functions. Therefore, the Greenfield Alternative analyses envelope the bounding scenarios for each of the modules that may be distributed to multiple sites and are applicable for the Distributed Greenfield Alternative. This conclusion is consistent with Section 2.10 of the AFCF Distributed Facility Study (WGI 2008b), which concluded that the accidents associated with each module should not differ from those of the Greenfield Alternative. Because each module has its own risk profile, the consequences and risks at any given site are dependent upon the specific module(s) located at that site; however, the overall accident consequences and risk will be enveloped by the consequences and risk at the bounding site. Use of the bounding site impacts is conservative, but this conservatism is offset by the duplication of functions such as waste management at multiple sites.

Distributed Brownfield Alternative-This alternative utilizes some combination of the existing facilities and new facilities at multiple sites (WGI 2008b). As with the Brownfield and Distributed Greenfield Alternatives, the Greenfield Alternative analyses provide scenarios that envelope each of the new or existing modules that may be distributed to multiple sites and are applicable for this alternative too. This conclusion is consistent with Section 2.10 of the AFCF Distributed Facility Study (WGI 2008b), which concluded that the accidents associated with each module should not differ from those of the Greenfield Alternative. Because each module has its own risk profile, the consequences and risks at any given site are dependent upon the specific modules located at that site; however, the overall accident consequences and risk will be enveloped by the consequences and risk at the bounding site. Use of the bounding site impacts is conservative, but this conservatism is offset by the duplication of functions such as waste management at multiple sites.

5. ACCIDENTS SELECTED FOR ANALYSIS

The methodology described in the previous section resulted in the selection of the accidents summarized in Table 5-1 for analysis. The accidents shown are applicable to all sites although some reflect unique site-specific conditions. The event frequency categories are based on frequencies for events in NEPA documents for similar facilities or other references as cited. There are no activities in nearby facilities that could initiate an accident in existing facilities used for these alternatives. Also, the accidents from the GNEP Program facilities are unlikely to initiate accidents at other facilities located at that site.

TABLE 5-1-*Accidents Selected for Analysis*

Accident Title	Frequency Category	Accident Initiator	Accident Phenomena	Comments
Radiological Accidents:				
Fuel Handling Accident	Anticipated (0.03/yr is used for this category)	• Internal • Natural phenomena	• Spill	Fuel or cask handling accidents have the potential to substantially impact workers, as demonstrated in several EISs.
Electrochemical Melter Eruption	Unlikely (10^{-3}/yr is used for this category)	• Internal	• Explosion	This is one of the bounding scenarios for electrochemical processing in the EMT EIS and SRS SNF EIS.
Explosion and Fire in Aqueous Separations	Unlikely (10^{-3}/yr is used for this category)	• Internal	• Explosion	This is one of the bounding scenarios in aqueous processing EISs.
Beyond Design Basis Earthquake	Extremely Unlikely (10^{-5}/yr is used for this category)	• Natural phenomena	• Earthquake	This is one of the bounding scenarios in the EISs reviewed. The magnitude of the earthquake is site specific and the capacity of existing facilities may differ from the capacity for new facilities.
Nuclear Criticality	Extremely Unlikely (10^{-5}/yr is used for this category)	• Internal • Natural phenomena	• Criticality	A nuclear criticality has the potential for bounding worker impacts.
Aircraft Crash	Beyond Extremely Unlikely (10^{-7}/yr since the frequency for AFCF will be no greater than the value for licensed reactors)	• External	• Fire • Spill	This is one of the bounding scenarios in several EISs reviewed.
Non-Radiological Accidents:				
Nitric Acid Release from Bulk Storage	Unlikely (10^{-3}/yr is used for this category)	• Internal • External • Natural phenomena	• Spill	This is one of the bounding chemical releases in at least one of the EISs reviewed and bounded other acid releases.

A textual description of each accident providing additional details and alternative-specific variations where appropriate follows. A scenario-specific table identifying the specific release parameters accompanies the description of each scenario. A basis is provided for each parameter.

5.1 Fuel Handling Accident

A fuel assembly or cask drop event can result in cladding failure and release of radioactive material from SNF. The SNF assembly or cask drop event can be the result of internal initiators such as operator error or equipment failure, or an external initiator such as an earthquake. In populated areas, SNF assemblies are only handled in robust shielded containers such as transportation casks, so an event involving a bare assembly in an occupied area is not credible. Transportation casks are designed to withstand the likely drop events and not expected to be damaged by a drop event. While there are many scenarios that cause minor damage to one or more fuel assemblies, the event analyzed is the drop of a fuel assembly during handling operations because the assembly may experience the maximum damage and release.

The fuel handling mishap is a 10 ft (3 m) free-fall of a single assembly. Ceramic fuels (e.g., LWR fuels) have a greater release fraction than metal fuels because ceramic matrices may fragment while metal fuel matrices would absorb energy by deforming but not fragment. Metal fuel would release primarily the gaseous elements in the clad gap. Therefore, the analysis bound the consequences of a drop by considering a ceramic LWR assembly. No credit is taken for the confinement of the fuel cladding, though even damaged cladding still provides considerable confinement. Credit is only taken for one stage of HEPA filtration even though there would be at least two stages. Inclusion of a second stage of HEPA filtration would reduce particulate releases by about two orders of magnitude (LANL 1986).

Given that there would be hundreds of fuel handling operations in the AFCF facility each year, the accident frequency category is estimated to be Anticipated. A noninvolved worker (NIW) and offsite individuals could be exposed to airborne radioactive material released after partial filtration through the ventilation system. Since fuel handling operations are performed in shielded cells with ventilation systems, facility workers would not be exposed to excess direct radiation or radioactive material. The release parameters used to analyze the consequences of this accident are presented in Table 5.1-1 along with a basis for the values used.

TABLE 5.1-1-*Release Parameters for the Fuel Handling Accident*

Parameter	Value	Basis/Comment
Release Point	Ground level	The event is conservatively assumed to occur with the doors open, which maximizes nearby impacts.
Duration	1 minute	A short duration release is conservatively assumed to ensure all receptors are present for the entire release.
MAR	1 LWR assembly (AFCF NEPA Data Study (WGI 2008a), Appendix A-3, Ci/MTIHM column adjusted to one assembly)	The inventory values in Appendix A-3 of the AFCF NEPA Data Study (WGI 2008a) are converted to assembly inventory values by multiplying by 0.5 MTIHM per assembly based on Section 2.2.2 of the AFCF NEPA Data Study (WGI 2008a).
DR	1	Assuming the entire MAR is involved is bounding.
ARF	1 volatiles[a] $7x10^{-5}$ particulates	All volatiles in the cladding gap could be released from failed fuel. The ARF x RF for particulates is based on Equation (4-1) of DOE-HDBK-3010-94 (DOE 2000i) using a 10-foot (3 m) drop height. The energy absorbing effects of the assembly structure and the partial confining effects of damaged cladding are not included in the analysis.
RF	included in the ARF	This factor is included with the ARF value above.
LPF	1 gases $1x10^{-3}$ particulates	This value is based on item (a) for the 1st stage of HEPA filtration in Table IX of LA-10294-MS (LANL 1986).

[a] The only radioactive gases present would be Kr-85 and I-129

5.2 Electrochemical Melter Eruption

The postulated electrochemical melter eruption event results from a buildup or addition of impurities in the melt. Impurities range from water which could cause a steam explosion to chemical contaminants which could cause a high-temperature exothermic reaction. As a result of the reaction in the melt, molten material is ejected from the melter into the processing structure. Cooling water pipes, if present within the process area, could be ruptured as a result of contact with the ejected material. An actual facility design would likely eliminate water pipes in the vicinity of the melter. Water released would be converted to steam, which could overwhelm the vessel and cell ventilation systems. Although the vessel and cell ventilation systems may be overwhelmed, there would be insufficient energy in the explosion to damage the facility structure or the larger-capacity final HEPA filtration system. Therefore, the melter eruption is assumed to occur with a single stage of HEPA filtration. The melter eruption is assumed to occur with a single stage of HEPA filtration even though there would be at least two stages. Inclusion of a second stage of HEPA filtration would reduce particulate releases by about two orders of magnitude (LANL 1986).

The NIW and offsite individuals could be exposed to airborne radioactive material released after partial filtration through the ventilation system, but facility workers are not expected to be directly exposed because facility walls are not damaged. The frequency category of this event is estimated to be Unlikely. The release parameters used to analyze the consequences of this accident are presented in Table 5.2-1 along with a basis for the values used.

TABLE 2.2-1-*Release Parameters for the Electrochemical Melter Eruption Accident*

Parameter	Value	Basis/Comment
Release Point	50 m stack	The event does not result in failure of the structure of stack, so the release is from the stack.
Duration	1 minute	A short duration release is conservatively assumed to ensure all receptors are present for entire release. Instantaneous
MAR	AFCF NEPA Data Study (WGI 2008a), Table 20, Ci/day column	The AFCF NEPA Data Study (WGI 2008a) considers both the oxide and metal fuels and reports the bound value.
DR	1	The event is conservatively assumed to involve the entire MAR.
ARF	1 volatiles[a] 0.2 Cs 1×10^{-3} other	See page F-26, Table F-18, EMT EIS (DOE 2000e).
RF	1	See page F-26, Table F-18, EMT EIS (DOE 2000e).
LPF	1 gases 1×10^{-3} particulates	This value is based on item (a) for the 1st stage of HEPA filtration in Table IX of LA-10294-MS (LANL 1986).

[a] The only radioactive gases present would be Kr-85 and I-129

5.3 Explosion and Fire in Aqueous Separations

A red oil explosion can occur when an organic solution, typically tri-n-butyl phosphate (TBP), and its diluents come in contact with concentrated nitric acid at a concentration greater than 10 moles/liter and a temperature above 130°C (266°F) without sufficient venting. Red oil is relatively stable below 130°C (266°F), but it can decompose explosively when its temperature is raised above 130°C (266°F). *Control of Red Oil Explosions in Defense Nuclear Facilities* (DNFSB 2003) provides additional details on the conditions and control measures for potential red oil explosions. A red oil explosion is possible in aqueous separations in equipment such as evaporators, acid concentrators, denitrators, and steam jets.

As a result of the reaction, the equipment ruptures and radioactive material is released to the cell. A fire involving the organic solution and its diluents could result from the event. The release could overwhelm the filtration system but is not expected to incapacitate the larger-capacity final HEPA system. There would be insufficient energy in the explosion to damage the facility structure, so facility workers would not be exposed to the release. Controls for prevention or mitigation of a red oil explosion may include controls on temperature, pressure, mass, and/or concentration.

The release phenomena could involve liquid sprays and a subsequent fire. After such an accident, the equipment contents are released and the final ventilation fans draw the airborne materials through a single stage of HEPA filtration. The NIW and offsite individuals could be exposed to airborne radioactive material released after partial filtration through the ventilation system, but facility workers are not expected to be directly exposed because facility walls are not damaged. The frequency category of this event is estimated to be Unlikely. The release parameters used to analyze the consequences of this accident are presented in Table 5.3-1 along with a basis for the values used.

TABLE 5.3-1-*Release Parameters for the Aqueous Separations Explosion and Fire Accident*

Parameter	Value		Basis/Comment
Release Point	50 m stack		The event does not result in failure of the building structure or stack, so the release is from the stack.
Duration	1 minute		The explosion is an instantaneous event and a resulting fire could occur promptly, so a short duration release model is appropriate.
MAR	AFCF NEPA Data Study (WGI 2008a), Appendix A-3, Ci/day column		The bounding batch size is assumed to be the same as the daily process rate. The MAR includes all radionuclides in the inventory even though some radionuclides are removed prior to some partitioning stages.
DR	1		The event is conservatively assumed to involve the entire MAR.
ARF	1 0.01	volatiles[a] non-volatiles	The values for volatiles and nonvolatiles are based on organic fires as reported in Sections 3.3.1 and 3.3.2 of DOE-HDBK-3010-94 (DOE 2000i).
RF	1		Assuming the entire release is respirable is bounding.
LPF	1 1×10^{-3}	gases particulates	This value is based on item (a) for the 1st stage of HEPA filtration in Table IX of LA-10294-MS (LANL 1986).

[a] The only radioactive gases present would be Kr-85 and I-129.

5.4 Beyond Design Basis Earthquake

The methodology for establishing DOE facility design criteria are presented in *Natural Phenomena Hazards Performance Categorization Guidelines for Structures, Systems, and Components*, DOE STD-1021-93 (DOE 2002f) and *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*, DOE STD-1020-2002 (DOE 2002g). The AFCF is expected to be a performance category 3 (PC-3) facility, which corresponds to $1\text{x}10^{-4}$ permissible annual probability of unacceptable performance [see Table C-1 of DOE STD-1020-2002 (DOE 2002g)]. Therefore, an earthquake capable of resulting in unacceptable performance (i.e., a beyond design basis earthquake) is expected to have a frequency of less than 1×10^{-4}/yr and thus fall within the frequency category of Extremely Unlikely. Component damage or malfunction and minor damage to the structure may occur for earthquakes somewhat beyond the design basis. Due to the safety factors used in structural design, gross structural failure or stack collapse is only expected for earthquakes with accelerations well beyond the design basis. Consistent with Section 3.2 of DOE's *Recommendations for Analyzing Accidents under the National Environmental Policy Act* (DOE 2002d), earthquakes of a magnitude capable of resulting in gross structural failure or stack collapse are not considered here since the consequences resulting from the facilities would be overwhelmed by and indistinguishable from the consequences unrelated to the AFCF. A stack collapse, were it to occur, would not significantly affect the offsite population impacts and is expected to result in a small increase in NIW impacts. The median frequency of a safe shutdown earthquake for current LWRs is $1.0x10^{-5}$/yr per Regulatory Guide 1.165 (NRC 1997k). The scenario selected for analysis here is an Extremely Unlikely event with an assumed frequency of 10^{-5} per year consistent with evaluations of commercial power reactors.

A beyond design basis earthquake may cause equipment malfunctions and result in a variety of events. The AFCF would have a robust, non-flammable facility design with combustible loading controls, so a facility-wide fire is not credible. An earthquake has the potential to damage the ventilation system and produce cracks in the cell enclosure, thereby resulting in a partially mitigated release. More severe events that result in damage to the confinement boundary or stack may increase consequences to nearby receptors but would have minimal effect on the population. The most impacting event of the more likely frequency categories is the explosion and fire in the aqueous separation process, so the bounding beyond design basis event is the explosion and fire in the aqueous process in conjunction with a compromise in the confinement boundary that results in an leak path factor (LPF) mid-way between total failure and intact performance of the HEPA filters.

One or more facility workers could be killed as a direct result of the earthquake, for example from falling debris. The NIW and the public may be exposed to the release. No credit is taken for the fire suppression efforts and equipment since the earthquake could incapacitate them. The magnitude of the earthquake is site specific. The accident frequency category is estimated to be Extremely Unlikely. The release parameters used to analyze the consequences of this accident are presented in Table 5.4-1 along with the basis for using these values.

TABLE 5.4-1-*Release Parameters for the Beyond Design Basis Earthquake Accident*

Parameter	Value		Basis/Comment
Release Point	50 m stack		The event does not result in failure of the building structure or stack, so the release is from the stack.
Duration	1 minute		The explosion is an instantaneous event and a resulting fire could occur promptly. A short duration release model is used, which assumes the majority of the release occurs from the explosion.
MAR	AFCF NEPA Data Study (WGI 2008a), Appendix A-3, Ci/day column		The bounding batch size is assumed to be the same as the daily process rate. The MAR includes all radionuclides in the inventory even though some radionuclides are removed prior to some partitioning stages.
DR	1		The event is conservatively assumed to involve the entire MAR.
ARF	1 0.01	volatiles[a] non-volatiles	The values for volatiles and nonvolatiles are based on organic fires as reported in Sections 3.3.1 and 3.3.2 of DOE-HDBK-3010-94 (DOE 2000i).
RF	1		Assuming the entire release is respirable is bounding.
LPF	1 0.03	gases particulates	This value reflects the degraded filtration system condition and is based on the geometric mean of 1 (complete release) and item (a) for the 1st stage of HEPA filtration in Table IX of LA-10294-MS (LANL 1986).

[a] The only radioactive gases present would be Kr-85 and I-129

5.5 Nuclear Criticality

An inadvertent nuclear criticality is possible in a facility such as the AFCF that contains substantial quantities of fissile material in various forms including spent nuclear fuel, solutions, powders, solids, and unirradiated nuclear fuel. A nuclear criticality can result if the quantity, concentration, configuration, moderation, or reflection of the fissile material sufficiently exceeds the criticality limits. The criticality limits could be violated due to initiators such as operator errors, equipment failures, process upsets, or a seismic event. A few examples of the types of criticality events that are possible include collapse of a storage vault/rack due to an earthquake, process upsets that result in concentration of fissile solutions in a process vessel, and operator error resulting in addition of moderator (e.g., water) to a product storage vault.

A criticality involving dissolved spent fuel is assumed to be the bounding AFCF criticality event because: 1) solution events are considered more likely and have a large number of fissions, 2) solid fissile forms retain pre-existing and generated fission products much more effectively than aqueous solutions, and 3) unirradiated materials do not contain pre-existing fission products, which may also be released in the event. The criticality event is assumed to involve 1×10^{19} total fissions, which results in a maximum evaporation of 26 gal (100 L) of solution (DOE 2000i). Events of this type are frequently modeled as an initial fission burst followed by smaller excursions over an eight-hour period [e.g., see DOE-HDBK-3010-94 (DOE 2000i) Section 6.1], but for simplicity, the event is assumed to result in a uniform release over a one-hour period in this analysis. It is assumed that some filters fail as a result of the event. The criticality event does not involve an abrupt energy release sufficient to fail multiple banks of HEPA filtration. A single stage of HEPA filtration is assumed to filter the release even though there would be at least two stages. Inclusion of a second stage of HEPA filtration would reduce particulate releases by about two orders of magnitude (LANL 1986)..

The NIW and offsite individuals could be exposed to a dose from inhalation and immersion in the plume of released nuclides from this event. Facility workers are not expected to be directly exposed to the release because facility walls are not damaged. Operations involving spent fuel solutions are performed behind shielding walls and the event would be promptly alarmed, so the increased direct radiation exposure to facility workers is not expected to be lethal. The release parameters used to analyze the consequences of this accident are presented in Table 5.5-1 along with a basis for the values used. The estimated frequency category of a criticality is Extremely Unlikely.

TABLE 5.5-1-*Release Parameters for the Nuclear Criticality Accident*

Parameter	Value	Basis/Comment
Release Point	50 m stack	The event does not result in failure of the building structure or stack, so the release is from the stack.
Duration	1 hour	The release is assumed to be uniform over a 1-hour perio Section 6.1 of DOE-HDBK-3010-94 (DOE 2000i) uses an initial burst with smaller subsequent excursions over an 8-hour perio This 1-hour release assumption simplifies the analysis and is more conservative.
MAR	DOE-HDBK-3010-94 (DOE 2000i) Table 6-7	This reference is applicable since it is for spent fuel solutions.
	See AFCF NEPA Data Study (WGI 2008a), Appendix A-3, Ci/day column	DOE-HDBK-3010-94 (DOE 2000i) Section 6.1 provides a basis for assuming release from 100 L of solution. The concentration of radionuclides is dependent upon the process stage involved, so this analysis is based conservatively on the daily throughput.
DR	1	Assuming the entire MAR is involved is bounding.
ARF	1 noble gas 0.25 iodine 1×10^{-3} ruthenium 5×10^{-4} other	These values are consistent with the values used in Section 6.3.1 of DOE-HDBK-3010-94 (DOE 2000i).
RF	1	Assuming the entire release is respirable is bounding.
LPF	1 gases 1×10^{-3} particulates	This value is based on item (a) for the 1st stage of HEPA filtration in Table IX of LA-10294-MS (LANL 1986).

5.6 Aircraft Crash

This scenario involves an aircraft crashing into the AFCF resulting in a breach of confinement. The crash could damage engineered barriers and may result in a criticality, fire, or spill event. Because of the robustness of the facility, there are a limited number of aircraft types capable of penetrating the shielding walls. Because of the very low likelihood of a penetrating crash and the small conditional probability that the event would be aligned to penetrate multiple cell walls, it is not credible that the crash would affect multiple processes (e.g., both the electrochemical and aqueous separation processes). The aqueous separation process is selected as the bounding location for the crash rather than the electrochemical separations process because it has a greater inventory and the release potential is at least as great. Each of the aqueous process steps contains fuel in a vulnerable form so all aqueous processes are vulnerable. The head-end process includes the voloxidation step, which transforms the fuel into a highly dispersible and respirable particulate form, and the dissolution step. The release fraction for an aircraft crash is based on release from the voloxidation process, but the release fraction from the aqueous separation product would be similar based on Table II of *Accident Analysis for Aircraft Crash Into Hazardous Facilities*, DOE-STD-3014-96 (DOE 2006q).

One or more facility workers could be killed as a direct result of the crash. The NIW and the public may be exposed to the release. The release would not be filtered since the facility confinement barrier is breached. No credit is taken for the mitigating effects of fire suppression efforts and equipment. The frequency category is estimated to be Beyond Extremely Unlikely but the specific likelihood is dependent upon site-specific factors such as proximity to airports and major flight paths. The siting of AFCF is expected to meet criteria similar to the NRC Aircraft Hazards criteria (NRC 2007l), therefore, an event frequency of 10^{-7} is used in this analysis. The release parameters used to analyze the consequences of this accident are presented in Table 5.6-1 along with a basis for the values used.

TABLE 5.6-1-*Release Parameters for the Aircraft Crash Accident*

Parameter	Value	Basis/Comment
Release Point	Ground level	Because the confinement barrier is breached, the release point could be at ground level.
Duration	1 minute	The release could occur over a short duration, so a short duration release model is appropriate.
MAR	AFCF NEPA Data Study (WGI 2008a), Appendix A-3, Ci/day column	Table A-3 of the AFCF NEPA Data Study (WGI 2008a) provides the bounding daily throughput, which is assumed to be the bounding inventory.
DR	1	Assuming the entire MAR is involved is bounding.
ARF	1 gases $2x10^{-3}$ particulates	Table II of *Accident Analysis for Aircraft Crash Into Hazardous Facilities*, DOE-STD-3014-96 (DOE 2006q) provides this value for evaluation of powder or aqueous liquid releases from aircraft crashes.
RF	1	Assuming the entire release is respirable is bounding.
LPF	1	Assuming all airborne material is released is bounding.

5.7 Nitric Acid Release from Bulk Storage

The AFCF would utilize a variety of hazardous chemicals in significant quantities. An accidental release of nitric acid from bulk storage is postulated as the bounding hazardous chemical event. Nitric acid is corrosive and can cause severe burns to all parts of the body. Its vapors are corrosive to the respiratory tract and may cause pulmonary edema which could prove fatal.

The leak could be the result of equipment failure, mechanical impact, or human error. The bulk storage building has precautions such as secondary confinement to mitigate the consequences of a nitric acid spill. However, it is possible for a spill associated with a delivery truck to occur where these precautions are not available.

The maximum storage of bulk chemicals is assumed to be equal to their annual usage. The annual usage of nitric acid is 1.6 $x10^6$ gal (5.9 x 10^6 L) of nitric acid per Table 11 of the AFCF NEPA Data Study (WGI 2008a). However, the consequence of this event is less dependent upon the volume of nitric acid spilled than on the surface area and temperature of the resulting pool. The bounding event is assumed to be an outdoor spill of nitric acid sufficient to result in a 1.1 x 10^4 ft^2 (1,000 m^2) pool of nitric acid with ambient and acid temperatures of 32° C (90° F). The nitric acid evaporates and is transported by the wind to all receptors. The DOE Protective Action Criteria, 60-minute AEGL-2 and 3 for nitric acid, are 24 and 92 ppm (SCAPA 2007). The estimated frequency category of this accident is estimated to be Unlikely.

6. CONSEQUENCE ANALYSIS METHODOLOGY

Accidents involving the release of radioactive or chemically hazardous materials place workers, members of the public, and the environment at risk. This section addresses the methodology used to estimate the consequences of each accident scenario analyzed.
Workers in the facility where the accident occurs may be particularly vulnerable to the affects of the accident because of their proximity. Consequence prediction becomes increasingly difficult to quantify for facility workers as the distance between the accident location and the worker decreases. This is because of the increased sensitivity and uncertainty associated with location and conditions. Factors such as shielding and air flow patterns can be considerably different for minor changes in location. The worker also may be injured or killed by physical effects of the accident itself. Therefore, worker impacts are addressed qualitatively and consequences for receptors less than 100 m from the point of release are not calculated.

Workers at least 100 m from the release point and the offsite public are also at risk of exposure to the extent that meteorological conditions transport the released materials in their direction. Using approved computer models, the dispersion of released radioactive and chemically hazardous materials and their effects are predicted. This section focuses on the consequence analysis methodology for these potential receptors.

In general, radiological doses that are unlikely to affect humans (e.g., doses below human radiation protection limits) are not known to cause measurable adverse effects to populations of plants and animals. Therefore, effects on the environment are addressed in qualitative terms based upon the impacts to humans.

6.1 Computer Codes

The consequence analysis uses computer codes for radioactive material releases and chemically hazardous material releases. The following subsections discuss the codes used for radioactive and chemically hazardous material releases.

6.1.1 Radioactive Material Releases

A deterministic, nonprobabilistic approach was used to analyze the consequences of the accident scenarios. The wide range of postulated accidents characterizes the range of impacts associated with the operation of the facilities being considered. The postulated accident scenario for radioactive material can be reasonably evaluated in terms of the effective dose equivalent, and from this, the bounding scenario can be determined.

The consequences of radioactive material releases are calculated using an atmospheric dispersion and radiological consequence calculation computer code. The code calculations incorporate scenario specific nuclide releases and conditions, such as release height and release duration. Site specific parameters (meteorology, distances to receptors, and surrounding population) are also incorporated.

Consequences of accidental radiological releases were determined using version 1.13.1 of the MACCS2 computer code (Chanin and Young 1998). MACCS2 is a United States Department of Energy/Nuclear Regulatory Commission (DOE/NRC) sponsored computer code that has been widely used in support of probabilistic risk assessments for the nuclear power industry and in support of safety and NEPA documentation for facilities throughout the DOE complex. The code meets DOE safety software assurance requirements (DOE 2007w).

The MACCS2 code uses three distinct modules for consequence calculations: ATMOS, EARLY, and CHRONIC. The ATMOS module performs atmospheric transport calculations, including dispersion, deposition, and decay. A straight-line Gaussian plume model is applied, with each hour's transport governed by the meteorology during that hour. Multiple calculations are performed for each release that include all sequential hourly meteorological conditions throughout the year. The EARLY module performs exposure calculations corresponding to the period immediately following the release; this module also includes the capability to simulate evacuation from areas surrounding the release. The EARLY module exposure pathways include inhalation, cloudshine (external exposure from the passing atmospheric plume), and groundshine (external exposure from nuclides deposited on the ground by the atmospheric plume). The CHRONC module considers the time period following the early phase; i.e., after the plume has passed. CHRONC exposure pathways include groundshine, resuspension inhalation, and ingestion of contaminated food and water. Land use interdiction (e.g., decontamination) can be simulated in this module. Other supporting input files include a meteorological data file and a site data file containing distributions of the population and agriculture surrounding the release site (Chanin and Young 1998). Jow et al. 1990 present a more detailed description of the model's methodology (Jow et al. 1990).

Because of the conservativeness of the assumptions used in this PEIS analysis, not all of the code's capabilities were used. For example, it was conservatively assumed that there would be no evacuation or protection of the surrounding population following an accidental release of radionuclides. Another conservative assumption was that wet and dry depositions of all radioactive material were set to zero for individual receptors [maximally exposed individual (MEI) and NIW]. These receptors are exposed for the duration of the release; suppressing deposition increases inhalation and cloudshine dose (increasing negative health effects) by keeping the radioactive material airborne (rather than depleting the plume by deposition) and available for inhalation. Deposition was also zero for population impact analyses. This assumption results in maximizing exposure to the release; long-term exposure pathways were not considered. Figure 6.1.1-1 illustrates the release and exposure pathways modeled in this analysis.

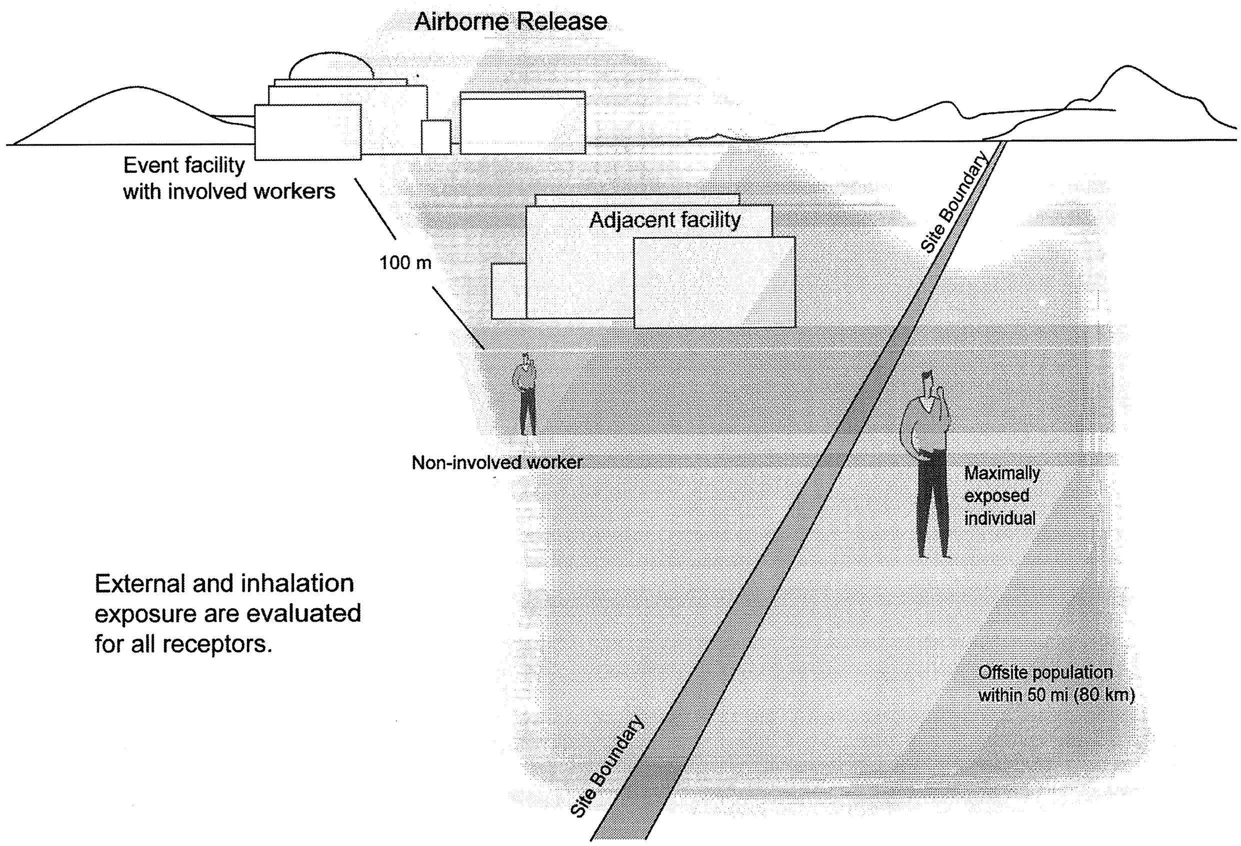

FIGURE 6.1.1-1—*Release and Exposure Pathways*

The meteorological data consisted of sequential hourly wind speed, wind direction, stability class and precipitation measured for 1 year. Ten radial rings and 16 uniform direction sectors were used to calculate the collective dose to the offsite population. The radial rings were every mile from 1 to 5 mi (2 to 8 km), a ring at 10 mi (16 km), and a ring every 10 mi (16 km), from 10 to 50 mi (16 to 80 km) starting at the distribution center. The location of the offsite MEI was assumed to be along the site boundary or, for elevated or buoyant releases, at the point of greatest offsite consequence. Similarly, the NIW location was taken as 328 ft (100 m) from the release in any direction.

MEI and NIW doses were calculated using conservative assumptions, such as the wind always blowing toward those receptors, locating the receptor along the plume centerline, and taking the distance to the MEI as the shortest distance in any direction to the site boundary. The doses (50-year committed effective dose equivalent) were converted into latent cancer fatalities (LCFs) using the factor of 6×10^{-4} LCFs per person-rem for both members of the public and workers (DOE 2002h). This factor was doubled for individual (MEI and NIW) receptors exposed to doses > 20 rem (DOE 2002h). Members of the public and workers are assumed to be exposed for the duration of the release; they or DOE would take protective or mitigative actions thereafter if required by the size of the release. Table 6.1.1-1 presents some MACCS2 parameter values that were used in the analysis (Chanin and Young 1998). To calculate the increased risk or likelihood of an LCF, an estimate of the accident probability, expressed as a frequency, must be known (i.e., Risk = Radiation Dose x LCF/Dose x Event Probability).

LATENT CANCER FATALITIES

As used in this PEIS, a latent cancer fatality (LCF) is a death resulting from cancer that has been caused by exposure to ionizing radiation. There is typically a latent period between the time of radiation exposure and the time the cancer cells become active. Exposure to radiation that results in a 1-rem lifetime dose causes an estimated 0.0006 chance of incurring a fatal cancer. In a population of 10,000 people, national statistics indicate that about 2,224 people would die from cancer of one form or another. Using information developed by the International Commission on Radiological Protection (ICRP 1991), if all 10,000 people received a dose of 0.167 rem during their lifetimes (in addition to normal background radiation dose), an estimated 1 additional cancer fatality would occur in that population. However, one would not be able to tell which of the 2,225 fatal cancers was caused by radiation and, possibly, the additional radiation would cause no fatal cancers.

Sometimes, calculations of the number of LCFs associated with radiation exposure do not yield whole numbers, and, especially in environmental applications, may yield numbers less than 1.0. For example, if each individual in a population of 100,000 received a total dose of 0.001 rem, the collective dose would be 100 person-rem and the corresponding estimated number of LCFs would be 0.06 (100,000 persons × 0.001 rem × 0.0006 LCFs per person-rem). This raises the issue: how should one interpret a non-integral number of LCFs, such as 0.06? The answer is to interpret the result as a statistical estimate. That is, 0.06 is the average number of deaths that would result if the same exposure situation were applied to many different groups of 100,000 people. For most groups, no one would incur a LCF from the 0.001 rem dose each member would have receive In a small fraction of the groups, 1 LCF would result; in exceptionally few groups, 2 or more latent fatal cancers would occur. The average number of deaths over all of the groups would be 0.06 LCFs (just as the average of 0, 0, 0, and 1 is 1/4, or 0.25). The most likely outcome for any single group is 0 LCFs.

TABLE 6.1.1-1—*General MACCS2 Analysis Assumptions*

Parameter	Selection	Comments
MACCS2		**Version 1.13.1**
Population	SECPOP2000 (NRC 2003) 1990 and 2000 census general population distributions extrapolated to 2060. Centered at accident source facility.	See topical reports for further discussion of extrapolation methodology.
Population Ring Boundaries	1, 2, 3, 4, 5, 10, 20, 30, 40, 50 mi (80 km)	General population to 50 mi (80 km).
Inhalation and external exposure from plume	Yes	
Inhalation and external exposure from deposition and resuspension	No	Deposition turned off to maximize downwind plume concentrations.
Breathing rate	16 in^3 (2.66×10^{-4} m^3) per second	Normal breathing rate, Chanin and Young 1998.
Evacuation	No	Assume no protective actions taken.
Relocation	No	Assume no protective actions taken.
Cloud shielding factor	0.75	Chanin and Young 1998.
Protection factor for inhalation	0.41	Chanin and Young 1998.
Skin protection factor	0.41	Chanin and Young 1998.
Ground shielding factor	0.33	Chanin and Young 1998. No deposition.
Wet deposition	No	No wet deposition, maximize downwind plume concentrations.

TABLE 6.1.1-1—*General MACCS2 Analysis Assumptions (continued)*

Parameter	Selection	Comments
Dry deposition	No	No dry deposition, maximize downwind plume concentrations.
Sigma-y, Sigma-z (dispersion parameters)	Tadmor-Gur Tables	Chanin and Young 1998.
Surface roughness length correction	1.27 (general population), 2.02 (MEI and NIW)	Corresponds to z0=10 centimeters (rural) for general population and z0=100 centimeters (urban) for individuals.
Plume meander time base	600 seconds	Chanin and Young 1998.
xpfac1	0.2	Plume meander exponential factor for time less than break point (1 hour). Chanin and Young 1998.
xpfac2	0.25	Chanin and Young 1998; plume meander exponential factor for times greater than 1 hour.
Plume segment reference time	0.5	Plume segment reference at center of release segment (for dispersion, deposition, decay calculations).
Atmospheric mixing height	Seasonal afternoon range (in 100's of meters): Hanford (6-20), INL (7-29), LANL (15-40), ORR (7 constant), SRS (12-19)	Site data and Holzworth 1972.
Wind shift without rotation	Yes	Plume direction follows wind direction every hour.
metcod	5	Stratified random samples for each day of the year (see nsmpls in the row below).
nsmpls	24	24 Meteorology samples per day (sample each hour).
Boundary conditions used in last ring	No	Hourly meteorology applied throughout model domain.
Dose conversion factors	FGR 11,12	
Presented dose results	TEDE-mean	
Health risk	6×10^{-4}	Fatal cancers per rem (total effective dose equivalent) (DOE 2002h). 1.2×10^{-3} for individuals exposed to doses greater than 20 rem.

Population and individual doses (MEI and NIW) were statistically sampled by assuming an equally likely accident start time during any hour of the year. All hours were sampled. The results from each of these samples were then incorporated into the mean results which are presented in this PEIS.

The impacts on an additional individual who is in the immediate vicinity of an accident, the involved worker who works at the facility where the accident is hypothesized to occur, are calculated using different methods than for the receptors described in Section 6.2.

The assumptions used in the GNEP PEIS consequence calculations may differ from the assumptions used in the NEPA documents used as inputs for scenarios selection; however, the GNEP PEIS analyses used the same assumptions throughout, thereby ensuring comparability of results reported here. As a result, the GNEP PEIS results can be compared directly with each other, though direct comparison with the

NEPA documents used as input may not be appropriate. The consistent, sometimes simplified, assumptions used in the GNEP PEIS are appropriate for the high-level programmatic comparisons in the GNEP PEIS.

6.1.2 Chemically Hazardous Material Releases

The consequences of accidental releases of hazardous chemicals were calculated using the Areal Location of Hazardous Atmospheres (ALOHA) code, version 5.4.1 (EPA 2007d). ALOHA is an EPA/National Oceanic and Atmospheric Administration (NOAA)-sponsored computer code that has been widely used in support of chemical accident responses and also in support of safety and NEPA documentation for DOE facilities. ALOHA is one of the codes designated by DOE's former Office of Environmental, Safety and Health as a toolbox code for safety analysis, as identified in *ALOHA Computer Code Application Guidance for Documented Safety Analysis Final Report* (DOE 2004h).

The ALOHA code is a deterministic representation of atmospheric releases of toxic and hazardous chemicals. The code can predict the rate at which chemical vapors escape (e.g., from puddles or leaking tanks) into the atmosphere; a specified release rate is also an option. In the case of the analyses performed here, the liquid chemical releases were determined based on the total chemical inventories, with ALOHA then predicting the chemical release rates from puddles.

Either of two dispersion algorithms is applied by the code, depending on whether the release is neutrally buoyant or heavier than air. The former is modeled similarly to radioactive releases in that the plume is assumed to advect (i.e., convey horizontally) with the wind velocity while dispersing laterally (horizontally perpendicular to the wind direction) and vertically. The latter considers the initial slumping and spreading of the release because of its density. As a heavier than air release becomes more dilute, its behavior tends towards that of a neutrally buoyant release.

The ALOHA code uses a constant set of meteorological conditions (e.g., wind speed, stability class) to determine the downwind atmospheric concentrations (EPA 2007d). Average conditions (mean wind speed and median stability class) were determined for each meteorological data set (see discussion of Radioactive Materials Release, above). This is roughly equivalent to the conditions corresponding to the mean radiological dose estimates of MACCS2 where the average results from hourly meteorological conditions were used. Accidental chemical release concentrations were calculated for the closest site boundary and at 100 meters (328 ft) from the release at each site.

ALOHA contains physical and toxicological properties for approximately 1,000 chemicals. The physical properties were used to determine which of the dispersion models and accompanying parameters were applied. Acute Exposure Guideline Levels, AEGL-2 and 3 (SCAPA 2007) are used to define the footprint of concern. Because the meteorological conditions specified do not account for wind direction (i.e., it is not known a priori in which direction the wind would be blowing in the event of an accident) the areas of concern are defined by a circle of radius equivalent to the downwind distance at which the concentration decreases to levels less than the level of concern.

6.2 Receptors

The impacts for the AFCF are evaluated at five DOE sites described:

- Hanford Site
- Idaho National Laboratory (INL)
- Los Alamos National Laboratory (LANL)

- Oak Ridge Reservation (ORR)
- Savannah River Site (SRS)

Accident consequences will be assessed for the following categories of people: involved workers, NIW, MEI, and the population.

6.2.1 Involved workers

The involved worker is a facility worker that is reasonably assumed to be either directly involved in the activity associated with the accident or in close proximity. Fatal or serious non-fatal injuries may be expected because of a worker's close proximity to the accident. Because the consequences are highly sensitive to assumptions regarding exact location, evacuation, etc., it is not credible to provide quantitative estimates of exposure for many scenarios. Therefore, the effects for involved workers will generally be described in semi-quantitative terms based on the likely number of people who would be involved and the general character of the accident scenario. Involved workers are assumed to evacuate the area 15 minutes after the event unless the event might reasonably incapacitate them.

6.2.2 Noninvolved worker

The NIW is a workers who would be on the site of the proposed action, but not involved in the action. The NIW will be defined as a person located 328 feet (100 meters) from the release or at the point of greatest impact when conditions mean this point is greater than 328 feet (100 meters) from the release. Elevated releases and releases involving plume rise may result in the point of greatest impact exceeding 328 feet (100 meters). The NIW may not be notified promptly after an accident so it is assumed that they are present for 1 hour, which generally exceeds the duration of the event.

6.2.3 Maximally Exposed Individual (MEI)

The accident analysis considers exposure to offsite individuals located anywhere outside of the site boundary or controlled access up to 50 miles from the AFCF location. The MEI is an individual located at a point at which the exposure to the accident scenarios would be a maximum of all of the offsite locations. The hypothetical MEI is taken to be located at or beyond the site boundary so that all offsite individuals will have an exposure no greater than this MEI. The dose to the MEI is calculated assuming the individual remains at that location throughout the duration of the accident; no protection actions are assumed for the MEI.

6.2.4 Population

Knowledge of the total number and the distribution of the offsite population around each of the proposed AFCF sites is required as input to the MACCS2 computer model, as discussed in Section 5.1.1.1. The collective dose to the public was calculated by considering only the off-site population within a 50-mile radius from the, site. A fifty-mile circular area is the standard range used in modeling consequences to the off-site population from an airborne release. The area was then divided into 16 pie-shaped wedges, each spanning 22.5-degree angles representing compass directions that start at north and move clockwise through north-northwest. The area was further divided into 10 annular regions, at radii corresponding to 1, 2, 3, 4, 5, 10, 20, 30, 40, and 50 miles from the center. The combination of 10 radial and 16 angular divisions resulted in 160 sectors in which the concentrations were calculated by the airborne dose models.

The population in each of these 160 sectors was calculated using 1990 and 2000 US Census population as provided in *SECPOP2000: Sector Population, Land Fraction, and Economic Estimation Program,* NUREG/CR-6525, Rev. 1 (NRC 2003). The population data are arrived at in essentially the following manner:

- The 160-sectors were overlain onto a regional map centered at the specified site coordinates.
- Block-group (BG) population data from the 1990 and 2000 U.S. Census for the 50-mile radius encompassing the site were considered. These data consist of total populations within the geographic boundaries in each BG.
- The geographic boundaries of each BG were defined overlain onto the sector map. Some sectors contained one or more whole BGs and/or partial BGs. BGs are attributed to the sector of their centroid location.
- The population in each sector was calculated as the sum of each BG's population.

The population was projected to the year 2060, corresponding to the endpoint of a 40-year operational period projected for an AFCF that initiates operation in the year 2020. The projection assumed a population change in each sector that was proportional to the change observed in that sector between 1990 and 2000. While most sectors experienced population growth between 1990 and 2000, some experienced population losses. Different projection models were applied to sectors experiencing population growth or population losses.

A "constant linear population growth" model was applied in cases where the population in a sector grew between 1990 and 2000. For the year 2060, the projected growth in each sector's population was calculated by taking the increase in each sector's populations between 1990 and 2000, multiplying this quantity by 6 and adding it to the year 2000 sector population. For example, if a sector's population increased from 100 to 110 people between 1990 and 2000, this 10-person growth was multiplied by 6 (i.e., 60 people) and added to the population of 110 for that sector in the year 2000. This resulted in a projected population of 170 people in that sector for the year 2060.

Application of the constant linear population growth model to a sector which experienced a population loss between 1990 and 2000 could possibly lead to a negative population when considering the long time over which such population changes are projected (see the discussion of alternate projection methods below). Therefore, a more conservative "constant population loss rate" model is used in such cases. For the year 2060 projection, any loss in a sector's population was calculated by taking the ratio of the sector's 2000 and 1990 populations, raising this ratio to the power of 6 and applying this scaling factor to the sector's year 2000 population. For example, if a sector's population decreased from 100 to 90 people between 1990 and 2000, the resulting ratio of 0.90 was raised to the power of 6 (scaling factor = 0.53). These scaling factors were applied to the year 2000 population of 90 in that sector. This resulted in a population projection of 48 people in that sector for the year 2060.

Population projections are fraught with uncertainty that only increases over time. Therefore, these projections should be considered in the context of establishing a common analytical framework for comparing impacts at alternative sites. Alternative population projection methods were considered, but were rejected in favor of the constant linear population growth and constant population loss rate models used here. These alternate methods included:

- The "constant population growth rate" model, which consists of calculating the rate of population growth by dividing the year 2000 population by the year 1990 population in each sector, and assuming this growth rate will remain constant over the projected period. This method would result in a significantly higher projected future population for those sectors where the population increased significantly between 1990 and 2000. Using the previous example, the projected population under this model for a sector that grew from 100 to 110 people between 1990 and 2000 would be 195.

- The "linear constant population loss" model, which consists of calculating the sector's population loss between 1990 and 2000, and multiplying the losses by 6 to the year 2060. The projected population losses would be significantly higher than losses estimated using the constant population loss rate model. Using the previous example, the projected populations under this model for a sector in which the population dropped from 100 to 90 people between 1990 and 2000 would be 30.

- The "national/state projection" model, which consists of taking official US Census Bureau (USCB) population projections for the United States and the state, or states, in which the 50-mile area for each site is located and applying the projected growth rates at specified annual intervals to each sector equally. The state populations are projected for the years 2000, 2005, 2010, 2015, and 2025 (USCB 2006a). The rate of change in the projected state populations would be calculated for each time interval and would be applied to the year 2000 population, projecting through the year 2025. The national population is projected in annual intervals from 2000 to 2100 (USCB 2002b). The rate of change from 2025 to 2060 can therefore be calculated. This rate of change would then be applied starting with the year 2025 population projection and continuing through 2060.

The alternative techniques are rejected because they do not appear to reduce uncertainty any better than the constant linear population growth and constant population loss rate models. Furthermore, an inspection of the national population projections indicated that the U.S. population is estimated to increase by an average of 25.7 million every 10 years from 2000 to 2050, ranging from a low of 24.6 million to a high of 26.3 million. This narrow range is consistent with a constant linear growth model through 2050. The direct application of the national and state projections, other than to validate the constant linear model, is also rejected. Use of the national/state projection model leads to a loss of resolution: any sector-specific population change that occurred between 1990 and 2000, data that are based on actual measured census BG data, are not considered in this model. Such a model does not detect population losses that might be occurring at the local level despite net population growth at the state or national level.

Table 6.2-1 provides the projected 50-mile population projections for 2060 used in this analysis.

TABLE 6.2-1-*Projected 2060 Population at DOE Sites*

Site	50-Mile Population in 2060
Hanford	8.5×10^5
Idaho National Laboratory	2.6×10^5
Los Alamos National Laboratory	9.4×10^5
Oak Ridge National Laboratory	2.0×10^6
Savannah River Site	1.4×10^6

6.3 Meteorological/Dispersion Parameters

Atmospheric dispersion of hypothetical accident releases at each site are calculated using an annual hourly sequential data set of meteorology measured at that site. Year to year variations in site specific meteorology are small; differences in atmospheric dispersion resulting from those variations are typically less than 10%. The met data set year chosen for each site was a recent year selected based on availability and completeness of the data set. At some sites (e.g., SRS, LANL) the data set was supplied directly in the form used by the MACCS2 model. At other sites (e.g., ORR) the data was reduced from a text file of hourly parameters. In the latter case, missing data was filled in either by interpolation (if the period was 4 hours or less) or substitution with the same site's met data from the same time period of the previous or subsequent year (for missing periods greater than 4 hours).

The relevant meteorological data for calculating health impacts from accidental releases of radionuclides are the wind speed and wind direction (at a height representative of the various accident scenario releases), the stability class, and the precipitation. The MACCS2 code requires input wind speed at 10-meters; the code calculates the vertical wind profile from this and determines the wind speed at the height of the plume. Table 6.3-1 contains the mean wind speed and the median stability class calculated from the sequential hourly meteorological data sets used as input to the accident impact calculations.

TABLE 6.3-1-*Summary of Meteorological Data Used in Accident Analysis*

Site (Year)	Tower Location	Mean Wind Speed at 10 meters height (meters/second)	Median Pasquill-Gifford Stability Class	Precipitation (inches)
Hanford (1998)	200 Area	2.90	D	Traces[a]
Idaho National Laboratory (2001)	INTEC	4.07	D	4.3
Los Alamos National Laboratory (2000)	Technical Area 54	2.63	D	14
Oak Ridge National Laboratory (2003)	ORR Met Tower B	1.23	E	65
Savannah River Site (2000)	H-Area	2.94	C	37

[a] Met data file indicates 42 hours of trace amounts of rainfall. Due to analysis deposition assumptions, this parameter is extraneous at all sites (see section 6.1.1.1).

A direction independent calculation of individual (MEI and NIW) exposure was performed. That is, the NIW, 100 meters from the hypothetical release, and the MEI, at the nearest site boundary from the facility, are always assumed to be downwind of the release, no matter the wind direction. The minimum MEI distance is indicated in Table 6.3-2.

TABLE 6.3-2-*Location of the Site Boundary*

Site	Distance from AFCF (miles)	Distance form AFCF (km)
Hanford	3.9	6.2
Idaho National Laboratory	8.4	13.5
Los Alamos National Laboratory	0.87	1.4
Oak Ridge National Laboratory	1.4	2.2
Savannah River Site	5.5	8.8

6.4 Agricultural Data

As discussed in Section 5.1.1.1, agricultural data are also inputs to MACCS2. Site-specific agricultural inputs are taken from *SECPOP2000: Sector Population, Land Fraction, and Economic Estimation Program,* NUREG/CR-6525, Rev. 1 (NRC 2003). The data are based on the 1997 U.S. Census of Agriculture. The length of the growing season and the fraction of farmland devoted to each crop type (e.g., pasture, grains, leafy vegetables, etc.) are specified for the 50 mile radius surrounding each site. Each of the 160-sectors defined in the population distribution is then associated with a data block in which the fraction of the land devoted to farming is specified. The fraction of each sector that is land is also specified. The methodology for creating the distribution of agricultural production around each site is analogous to that used to create the population distribution (see section 5.3.4), except that county farm production is used. Each sector is then associated with a county. Because of the uncertainty in each site's future agriculture production and the no deposition assumption (see Section 6.1.1.1) which negates exposure from this pathway, no attempt was made to extrapolate the recent agricultural production data into the future.

7. RISK AND CONSEQUENCE ANALYSIS RESULTS

This section presents the impacts of the bounding AFCF radiological and hazardous chemical accidents for the various receptors.

7.1 Radioactive Materials Release Accidents

Tables 7.1-1 through 7.1-5 present the impacts of bounding radiological accident scenarios for the AFCF at the specific DOE sites. The results are presented in terms of health impacts (risk of incremental LCFs) for the following receptors:

- The MEI is assumed located at the nearest site boundary or the point of greatest impact beyond the boundary,
- A hypothetical NIW is a site worker not directly involved with operation of the facility, but located 100 meters from the facility;
- The offsite population (projected to year 2060) within 50 miles of each site.

The results are reorganized and presented as the impact from all accidents at all sites to the Offsite Population (Table 7.1-6), MEI (Table 7.1-7), and the NIW (Table 7.1-8). The risk of an accident reflects Xthe probability or frequency of occurrence and is calculated by multiplying the accident's frequency by the accident's consequences. The increased annual risk of a latent cancer fatality from GNEP Program operations to the MEI would be less than 10^{-6}; the expected increased number of cancers in the surrounding population from GNEP Program operations would be 0 (i.e., $<10^{-4}$).

TABLE 7.1-1—*AFCF Accident Risks, Hanford*

Accident	MEI[a]	Offsite Population[b,c]	NIW[a]
Fuel Handling Accident	4.4E-09	6.1E-06	5.8E-07
Electrochemical Melter Eruption	4.2E-10	6.5E-07	2.0E-09
Explosion and Fire in Aqueous Separations	7.2E-09	1.1E-05	3.3E-08
Beyond Design Basis Earthquake	2.2E-09	3.3E-06	1.0E-08
Nuclear Criticality	4.2E-12	6.7E-09	3.6E-11
Aircraft Crash	1.3E-10	1.9E-07	5.2E-09

[a] Increased likelihood of a LCF per year of operation.
[b] Increased number of expected LCFs per year of operation.
[c] Based on a projected 2060 population of approximately 8.5 x 10^5 persons residing within 50 mi (80 km) of facility.

TABLE 7.1-2—*AFCF Accident Risks, INL*

Accident	MEI[a]	Offsite Population[b,c]	NIW[a]
Fuel Handling Accident	7.3E-10	2.8E-07	4.0E-07
Electrochemical Melter Eruption	7.8E-11	3.0E-08	7.1E-10
Explosion and Fire in Aqueous Separations	1.3E-09	5.2E-07	1.2E-08
Beyond Design Basis Earthquake	4.0E-10	1.5E-07	3.6E-09
Nuclear Criticality	7.7E-13	2.9E-10	2.1E-11
Aircraft Crash	2.2E-11	9.7E-09	7.4E-09

[a] Increased likelihood of a LCF per year of operation.
[b] Increased number of expected LCFs per year of operation.
[c] Based on a projected 2060 population of approximately 2.6 x 10^5 persons residing within 50 mi (80 km) of facility.

TABLE 7.1-3—*AFCF Accident Risks, LANL*

Accident	MEI[a]	Offsite Population[b,c]	NIW[a]
Fuel Handling Accident	3.9E-08	1.1E-05	6.0E-07
Electrochemical Melter Eruption	2.5E-09	9.8E-07	1.1E-09
Explosion and Fire in Aqueous Separations	4.2E-08	1.7E-05	1.9E-08
Beyond Design Basis Earthquake	1.2E-08	5.0E-06	5.8E-09
Nuclear Criticality	2.3E-11	1.0E-08	3.1E-11
Aircraft Crash	4.5E-10	2.7E-07	5.2E-09

[a] Increased likelihood of a LCF per year of operation.
[b] Increased number of expected LCFs per year of operation.
[c] Based on a projected 2060 population of approximately 9.4×10^5 persons residing within 50 mi (80 km) of facility.

TABLE 7.1-4—*AFCF Accident Risks, ORR*

Accident	MEI[a]	Offsite Population[b,c]	NIW[a]
Fuel Handling Accident	7.9E-08	3.9E-05	1.8E-06
Electrochemical Melter Eruption	4.9E-09	4.0E-06	6.8E-10
Explosion and Fire in Aqueous Separations	8.4E-08	6.8E-05	1.2E-08
Beyond Design Basis Earthquake	2.5E-08	2.0E-05	3.5E-09
Nuclear Criticality	4.5E-11	4.0E-08	5.0E-11
Aircraft Crash	7.3E-10	1.1E-06	4.8E-10

[a] Increased likelihood of a LCF per year of operation.
[b] Increased number of expected LCFs per year of operation.
[c] Based on a projected 2060 population of approximately 2.0×10^6 persons residing within 50 mi (80 km) of facility.

TABLE 7.1-5—*AFCF Accident Risks, SRS*

Accident	MEI[a]	Offsite Population[b,c]	NIW[a]
Fuel Handling Accident	8.6E-10	2.8E-06	3.1E-07
Electrochemical Melter Eruption	9.0E-11	3.1E-07	2.0E-09
Explosion and Fire in Aqueous Separations	1.5E-09	5.2E-06	3.4E-08
Beyond Design Basis Earthquake	4.6E-10	1.6E-06	1.0E-08
Nuclear Criticality	9.0E-13	3.2E-09	3.2E-11
Aircraft Crash	2.6E-11	9.8E-08	5.6E-09

[a] Increased likelihood of a LCF per year of operation.
[b] Increased number of expected LCFs per year of operation.
[c] Based on a projected 2060 population of approximately 1.4×10^6 persons residing within 50 m (80 km) of facility.

TABLE 7.1-6—*AFCF Accident Risks[a] to the Offsite Population (All Sites)*

Accident	Hanford	INL	LANL	ORR	SRS
Fuel Handling Accident	6.1E-06	2.8E-07	1.1E-05	3.9E-05	2.8E-06
Electrochemical Melter Eruption	6.5E-07	3.0E-08	9.8E-07	4.0E-06	3.1E-07
Explosion and Fire in Aqueous Separations	1.1E-05	5.2E-07	1.7E-05	6.8E-05	5.2E-06
Beyond Design Basis Earthquake	3.3E-06	1.5E-07	5.0E-06	2.0E-05	1.6E-06
Nuclear Criticality	6.7E-09	2.9E-10	1.0E-08	4.0E-08	3.2E-09
Aircraft Crash	1.9E-07	9.7E-09	2.7E-07	1.1E-06	9.8E-08

[a] Increased number of expected LCFs per year of operation.

The accident with the highest risk to the offsite population is the "Explosion and Fire in Aqueous Separations" scenario. The collective risk to the offsite population for this scenario would range from 5.2×10^{-7} expected LCFs per year of operation in the INL offsite population (approximately 2.6×10^5 persons) to 6.8×10^{-5} expected LCFs per year of operation in the ORR offsite population (approximately 2.0×10^6 people).

TABLE 7.1-7—*AFCF* Accident Risks[a] to the Maximally Exposed Individual (All Sites)

Accident	Hanford	INL	LANL	ORR	SRS
Fuel Handling Accident	4.4E-09	7.3E-10	3.9E-08	7.9E-08	8.6E-10
Electrochemical Melter Eruption	4.2E-10	7.8E-11	2.5E-09	4.9E-09	9.0E-11
Explosion and Fire in Aqueous Separations	7.2E-09	1.3E-09	4.2E-08	8.4E-08	1.5E-09
Beyond Design Basis Earthquake	2.2E-09	4.0E-10	1.2E-08	2.5E-08	4.6E-10
Nuclear Criticality	4.2E-12	7.7E-13	2.3E-11	4.5E-11	9.0E-13
Aircraft Crash	1.3E-10	2.2E-11	4.5E-10	7.3E-10	2.6E-11

[a] Increased likelihood of a LCF per year of operation.

For the **MEI**, the "Explosion and Fire in Aqueous Separations" scenario would result in an increased risk of a latent cancer fatality of 1.3×10^{-9} per year of operation (INL) to 8.4×10^{-8} per year of operation (ORR).

TABLE 7.1-8—*AFCF Accident Risks[a] to the Noninvolved Worker (All Sites)*

Accident	Hanford	INL	LANL	ORR	SRS
Fuel Handling Accident	5.8E-07	4.0E-07	6.0E-07	1.8E-06	3.1E-07
Electrochemical Melter Eruption	2.0E-09	7.1E-10	1.1E-09	6.8E-10	2.0E-09
Explosion and Fire in Aqueous Separations	3.3E-08	1.2E-08	1.9E-08	1.2E-08	3.4E-08
Beyond Design Basis Earthquake	1.0E-08	3.6E-09	5.8E-09	3.5E-09	1.0E-08
Nuclear Criticality	3.6E-11	2.1E-11	3.1E-11	5.0E-11	3.2E-11
Aircraft Crash	5.2E-09	7.4E-09	5.2E-09	4.8E-10	5.6E-09

[a] Increased likelihood of a LCF per year of operation.

The accident with the highest onsite risk is the "Fuel Handling Accident" scenario. The risk to the NIW would range from 3.1×10^{-7} at SRS to 1.8×10^{-6} at ORR.

Consequence impacts at each site, in terms of radiation dose and corresponding LCFs, are given in Tables 7.1-9 through 7.1-13. The same results are reorganized and presented as the impact from all accidents at all sites to the Offsite Population (Table 7.1-14), MEI (Table 7.1-15), and the NIW (Table 7.1-16). Consequences assume that the accident has occurred and, therefore, the probability or frequency of the accident is not taken into account.

TABLE 7.1-9—*AFCF Accident Consequences at Hanford*

Accident	MEI[a]		Offsite Population[b]		NIW	
	Dose (rem)	LCFs[c]	Dose (Person-rem)	LCFs[d]	Dose (rem)	LCFs[c]
Fuel Handling Accident	2.5E-04	1.5E-07	3.4E-01	2.0E-04	3.2E-02	1.9E-05
Electrochemical Melter Eruption	7.1E-04	4.2E-07	1.1E+00	6.5E-04	3.3E-03	2.0E-06
Explosion and Fire in Aqueous Separations	1.2E-02	7.2E-06	1.8E+01	1.1E-02	5.6E-02	3.3E-05
Beyond Design Basis Earthquake	3.6E-01	2.2E-04	5.5E+02	3.3E-01	1.7E+00	1.0E-03
Nuclear Criticality	7.0E-04	4.2E-07	1.1E+00	6.7E-04	6.0E-03	3.6E-06
Aircraft Crash	2.1E+00	1.3E-03	3.1E+03	1.9E+00	4.4E+01	5.2E-02

[a] At site boundary, approximately 3.9 mi (6.2 km) from release.
[b] Based on a projected 2060 population of approximately 8.5×10^5 persons residing within 50 mi (80 km) of AFCF location.
[c] Increased likelihood of a latent cancer fatality.
[d] Increased number of LCFs.

TABLE 7.1-10—*AFCF Accident Consequences at INL*

Accident	MEI[a]		Offsite Population[b]		NIW	
	Dose (rem)	LCFs[c]	Dose (Person-rem)	LCFs[d]	Dose (rem)	LCFs[c]
Fuel Handling Accident	4.1E-05	2.4E-08	1.5E-02	9.2E-06	2.2E-02	1.3E-05
Electrochemical Melter Eruption	1.3E-04	7.8E-08	5.1E-02	3.0E-05	1.2E-03	7.1E-07
Explosion and Fire in Aqueous Separations	2.2E-03	1.3E-06	8.6E-01	5.2E-04	2.0E-02	1.2E-05
Beyond Design Basis Earthquake	6.6E-02	4.0E-05	2.6E+01	1.5E-02	6.1E-01	3.6E-04
Nuclear Criticality	1.3E-04	7.7E-08	4.9E-02	2.9E-05	3.5E-03	2.1E-06
Aircraft Crash	3.7E-01	2.2E-04	1.6E+02	9.7E-02	6.2E+01	7.4E-02

[a] At site boundary, approximately 8.4 mi (13.5 km) from release.
[b] Based on a projected 2060 population of approximately 2.6×10^5 persons residing within 50 mi (80 km) of AFCF location.
[c] Increased likelihood of a latent cancer fatality.
[d] Increased number of LCF.

TABLE 7.1-11—*AFCF Accident Consequences at LANL*

Accident	MEI[a]		Offsite Population[b]		NIW	
	Dose (rem)	LCFs[c]	Dose (Person-rem)	LCFs[d]	Dose (rem)	LCFs[c]
Fuel Handling Accident	2.2E-03	1.3E-06	6.4E-01	3.8E-04	3.3E-02	2.0E-05
Electrochemical Melter Eruption	4.1E-03	2.4E-06	1.6E+00	9.8E-04	1.9E-03	1.1E-06
Explosion and Fire in Aqueous Separations	6.9E-02	4.2E-05	2.8E+01	1.7E-02	3.2E-02	1.9E-05
Beyond Design Basis Earthquake	2.1E+00	1.2E-03	8.4E+02	5.0E-01	9.7E-01	5.8E-04
Nuclear Criticality	3.9E-03	2.3E-06	1.7E+00	1.0E-03	5.2E-03	3.1E-06
Aircraft Crash	7.4E+00	4.5E-03	4.5E+03	2.7E+00	4.3E+01	5.2E-02

[b] At site boundary, approximately 0.87 mi (1.4 km) from release.
[c] Based on a projected 2060 population of approximately 9.4×10^5 persons residing within 50 mi (80 km) of AFCF location.
[d] Increased likelihood of a latent cancer fatality.
[e] Increased number of LCF.

TABLE 7.1-12—*AFCF Accident Consequences at ORR*

Accident	MEI[a]		Offsite Population[b]		NIW	
	Dose (rem)	LCFs[c]	Dose (Person-rem)	LCFs[d]	Dose (rem)	LCFs[c]
Fuel Handling Accident	4.4E-03	2.7E-06	2.2E+00	1.3E-03	1.0E-01	6.0E-05
Electrochemical Melter Eruption	8.2E-03	4.9E-06	6.7E+00	4.0E-03	1.1E-03	6.8E-07
Explosion and Fire in Aqueous Separations	1.4E-01	8.4E-05	1.1E+02	6.8E-02	1.9E-02	1.2E-05
Beyond Design Basis Earthquake	4.2E+00	2.5E-03	3.4E+03	2.0E+00	5.8E-01	3.5E-04
Nuclear Criticality	7.6E-03	4.5E-06	6.7E+00	4.0E-03	8.4E-03	5.0E-06
Aircraft Crash	1.2E+01	7.3E-03	1.8E+04	1.1E+01	7.9E+00	4.8E-03

[b] At site boundary, approximately 1.4 mi (2.2 km) from release.
[c] Based on a projected 2060 population of approximately 2.0×10^6 persons residing within 50 mi (80 km) of AFCF location.
[d] Increased likelihood of a latent cancer fatality.
[e] Increased number of LCF.

TABLE 7.1-13—*AFCF Accident Consequences at SRS*

Accident	MEI[a]		Offsite Population[b]		NIW	
	Dose (rem)	LCFs[c]	Dose (Person-rem)	LCFs[d]	Dose (rem)	LCFs[c]
Fuel Handling Accident	4.8E-05	2.9E-08	1.6E-01	9.4E-05	1.8E-02	1.1E-05
Electrochemical Melter Eruption	1.5E-04	9.0E-08	5.1E-01	3.1E-04	3.3E-03	2.0E-06
Explosion and Fire in Aqueous Separations	2.6E-03	1.5E-06	8.7E+00	5.2E-03	5.6E-02	3.4E-05
Beyond Design Basis Earthquake	7.7E-02	4.6E-05	2.6E+02	1.6E-01	1.7E+00	1.0E-03
Nuclear Criticality	1.5E-04	9.0E-08	5.4E-01	3.2E-04	5.3E-03	3.2E-06
Aircraft Crash	4.4E-01	2.6E-04	1.6E+03	9.8E-01	4.6E+01	5.6E-02

[b] At site boundary, approximately 5.5 mi (8.8 km) from release.
[c] Based on a projected 2060 population of approximately $1.4x10^6$ persons residing within 50 mi (80 km) of AFCF location.
[d] Increased likelihood of a latent cancer fatality.
[c] Increased number of latent cancer fatalities.

TABLE 7.1-14—*AFCF Accident Health Consequences (Increased Number of Latent Cancer Fatalities) to the Offsite Population (All Sites)*

Accident	Hanford	INL	LANL	ORR	SRS
Fuel Handling Accident	2.0E-04	9.2E-06	3.8E-04	1.3E-03	9.4E-05
Electrochemical Melter Eruption	6.5E-04	3.0E-05	9.8E-04	4.0E-03	3.1E-04
Explosion and Fire in Aqueous Separations	1.1E-02	5.2E-04	1.7E-02	6.8E-02	5.2E-03
Beyond Design Basis Earthquake	3.3E-01	1.5E-02	5.0E-01	2.0E+00	1.6E-01
Nuclear Criticality	6.7E-04	2.9E-05	1.0E-03	4.0E-03	3.2E-04
Aircraft Crash	1.9E+00	9.7E-02	2.7E+00	1.1E+01	9.8E-01

The accident with the highest consequence to the offsite population would be the Aircraft Crash scenario. Using the dose-to-risk conversion factor of $6x10^{-4}$ per person-rem the collective population dose in the Beyond Extremely Unlikely event that this accident were to occur is estimated to result in $9.7x10^{-2}$ additional LCFs at INL to 11 additional LCFs at ORR.

TABLE 7.1-15—*AFCF Accident Health Consequences (Increased Likelihood of a Latent Cancer Fatality) to the Maximally Exposed Individual (All Sites)*

Accident	Hanford	INL	LANL	ORR	SRS
Fuel Handling Accident	1.5E-07	2.4E-08	1.3E-06	2.7E-06	2.9E-08
Electrochemical Melter Eruption	4.2E-07	7.8E-08	2.4E-06	4.9E-06	9.0E-08
Explosion and Fire in Aqueous Separations	7.2E-06	1.3E-06	4.2E-05	8.4E-05	1.5E-06
Beyond Design Basis Earthquake	2.2E-04	4.0E-05	1.2E-03	2.5E-03	4.6E-05
Nuclear Criticality	4.2E-07	7.7E-08	2.3E-06	4.5E-06	9.0E-08
Aircraft Crash	1.3E-03	2.2E-04	4.5E-03	7.3E-03	2.6E-04

Using the dose-to-risk conversion factor of 6×10^{-4} per person-rem (or twice 6×10^{-4} per person-rem for individual doses greater than 20 rem), for the Beyond Extremely Unlikely Aircraft Crash accident, the INL MEI dose is estimated to result in an increased likelihood of latent cancer fatality of 2.2×10^{-4} and the ORR MEI dose is estimated to result in an increased likelihood of latent cancer fatality of 7.3×10^{-3}.

TABLE 7.1-16—*AFCF Accident Health Consequences (Increased Likelihood of a Latent Cancer Fatality) to the Noninvolved Worker (All Sites)*

Accident	Hanford	INL	LANL	ORR	SRS
Fuel Handling Accident	1.9E-05	1.3E-05	2.0E-05	6.0E-05	1.1E-05
Electrochemical Melter Eruption	2.0E-06	7.1E-07	1.1E-06	6.8E-07	2.0E-06
Explosion and Fire in Aqueous Separations	3.3E-05	1.2E-05	1.9E-05	1.2E-05	3.4E-05
Beyond Design Basis Earthquake	1.0E-03	3.6E-04	5.8E-04	3.5E-04	1.0E-03
Nuclear Criticality	3.6E-06	2.1E-06	3.1E-06	5.0E-06	3.2E-06
Aircraft Crash	5.2E-02	7.4E-02	5.2E-02	4.8E-03	5.6E-02

Using the dose-to-risk conversion factor from the previous paragraph, if the Beyond Extremely Unlikely Aircraft Crash event were to occur, the ORR NIW has a probability of 4.8×10^{-3} and the INL dose has a probability of 7.4×10^{-2} of resulting in the development of a LCF.

7.2 Hazardous Chemical Release Accidents

Table 7.2-1 presents the impacts of a release caused by a hypothetical spill of nitric acid at each of the five DOE sites. Evaporation from the pool of acid caused by the Unlikely spill would result in downwind airborne concentrations which can exceed DOE Protective Action Criteria (SCAPA 2007). As shown in the table, a NIW 330 ft (100 m) downwind of the spill at any of the sites would be exposed to levels well in excess of nitric acid's AEGL-3 concentration; life-threatening health effects up to death would likely occur. On the other hand, at all sites, the public downwind of the release would be subject to nitric acid concentrations levels below levels that would cause irreversible or serious health effects.

TABLE 7.2-1—*AFCF Nitric Acid Spill Impacts*

	Concentration at			
Site	Distance to AEGL-2[a] (ft)	Distance to AEGL-3[b] (ft)	NIW[c] (ppm)	MEI[d] (ppm)
Hanford	4.1×10^3	1.8×10^3	1.1×10^3	0.77
INL	2.2×10^3	1.0×10^3	430	0.44
LANL	4.1×10^3	1.7×10^3	1.2×10^3	15
ORR	3.7×10^3	1.6×10^3	1.5×10^3	4.9
SRS	3.3×10^3	1.6×10^3	1.0×10^3	2.9

[a] AEGL-2 concentration for nitric acid is 24 ppm. AEGL-2 is the airborne concentration above which it is predicted that the general population, including susceptible individuals, could experience irreversible or other serious, long-lasting adverse health effects or an impaired ability to escape.
[b] AEGL-3 concentration for nitric acid is 92 ppm. AEGL-3 is the airborne concentration above which it is predicted that the general population, including susceptible individuals, could experience life-threatening health effects or death.
[c] Located 328 ft (100 m) from the release.
[d] Located at the nearest site boundary to the release. The distances to the site boundary are 3.9 (Hanford), 8.4 (INL), 0.87 (LANL), 1.4 (ORR), and 5.5 (SRS) mi [6.2 (Hanford), 13.5 (INL), 1.4 (LANL), 2.2 (ORR), and 8.8 (SRS) km].

7.3 Involved Worker Impacts

Workers in the facility where the accident occurs would be particularly vulnerable to the effects of the accident because of their location. For all of the accidents, there is a potential for injury or death to involved workers in the vicinity of the accident. However, prediction of latent potential health effects becomes increasingly difficult to quantify for facility workers as the distance between the accident location and the worker decreases. This is because the individual worker exposure cannot be precisely defined with respect to the presence of shielding and other protective features. The worker also may be injured or killed by physical effects of the accident itself. Note that the potential for any of these consequences is greatly reduced by the use of shielding and remote operations.

The facility ventilation system would control dispersal of the airborne radiological debris from the accident. Following initiation of accident/site emergency alarms, workers would evacuate the area in accordance with site emergency operating procedures and would not be vulnerable to additional radiological injury.

The bounding case radiological accident for involved workers is an inadvertent criticality. Severe worker exposures could occur inside the facility as a result of a criticality, due primarily to the effects of prompt neutrons and gammas. A criticality would be detected by the criticality alarm system, and an evacuation alarm would be sounded. All personnel would immediately evacuate the building.

Personnel close to the criticality event (within the building) may incur prompt external exposures. Depending on distance and the amount of intervening shielding material, lethal doses composed of neutron and gamma radiation could be delivered. The dose due to prompt gamma and neutron radiation at a distance can be evaluated by the following formulas (DOE 2005n):

Prompt gamma dose : $D_g = 2.1\times10^{-20}\ N\ d^{-2}\ \exp^{-3.4d}$

Prompt neutron dose: $D_n = 7.0\times10^{-20}\ N\ d^{-2}\ \exp^{-5.2d}$

Where:

D_g = gamma dose (rem)

D_n = neutron dose (rem) (neutron quality factor = 20)

N = number of fissions

d = distance from source (km)

exp = the base of the natural logarithm

At a distance of 10 meters, the combined prompt gamma and neutron radiation dose to personnel from a criticality event (1×10^{19} fissions) would be 8.7×10^{3} rem (Dg = 2,030 rem plus Dn = 6,645 rem). A dose of approximately 450 rem received in a short period of time would result in death to 50 percent of the exposed population within 30 days if there is no medical intervention (DOE 1999e). Thus, the potential for lethal exposure exists. In reality, at-risk operations would be conducted remotely behind shielded walls, greatly mitigating these consequences.

The facility interior concrete walls would provide substantial shielding, except through the doors. In the event of a criticality, this shielding and rapid evacuation from the facility would reduce doses to personnel not in the immediate vicinity of the criticality excursion.

Direct exposure to airborne fission products produced during the criticality event would contribute only a small fraction to the total dose to a worker. Because of ventilation system operation, other personnel inside the building would not likely incur radiation dose resulting from the inhalation of airborne radioactive materials or immersion in the plume. If the ventilation system were unavailable, this dose would be small in comparison to the direct dose received at the time of the burst. The worker immediately involved would act appropriately according to training and emergency procedures.

8. REFERENCES

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Appendix A: Hazard Checklist

The hazard checklist presented in *Safety Analysis and Risk Assessment Handbook (SARAH)*, (RFETS 2002) was adapted for use in the AFCF accident analysis. Table A-1 presents the hazard checklist used for the AFCF accident analysis. Each hazard identified in the first column is reviewed to first identify whether or not it is relevant to AFCF (i.e., the second column) and whether or not the hazard is beyond a standard industrial hazard (SIH) (i.e., the third column). A hazard is a SIH if they are adequately addressed by DOE-prescribed programs and DOE or national consensus codes or standards. Standard industrial hazards do not warrant analysis and are dismissed. Hazards that are present and beyond SIHs are considered candidate accidents for the AFCF accident analysis.

Table A-1. Hazard Checklist.

	Hazard	Present?	Beyond SIH?	Comments
1.	**Electrical**			
1.1	Battery Banks	☒	☐	
1.2	Diesel Generators	☒	☐	
1.3	High-Voltage Lines	☒	☐	
1.4	Transformers	☒	☐	
1.5	Wiring	☒	☐	
1.6	Switchgear	☒	☐	
1.7	Underground Wiring	☒	☐	
1.8	Overhead Wiring	☒	☐	
1.9	Cable Runs	☒	☐	
1.10	Service Outlets and Fittings	☒	☐	
1.11	Pumps	☒	☐	
1.12	Motors	☒	☐	
1.13	Heaters	☒	☐	
1.14	Compressors	☒	☐	
1.15	Grounding Grids	☒	☐	
1.16	Lightning Grids	☒	☐	
2.	**Explosives-Electrophorics**			
2.1	Plutonium	☒	☒	SNF, recovered material, and fresh fuel.
2.2	Uranium	☒	☒	SNF, recovered material, and fresh fuel.
2.3	Sodium	☒	☐	
2.4	Potassium	☒	☐	
2.5	Dusts	☒	☐	
2.6	Hydrogen (bulk)	☒	☐	
2.7	Hydrogen (batteries)	☒	☐	
2.8	Other Flammable Gases	☒	☐	
2.9	Nitrates	☐	☐	
2.10	Peroxides	☐	☐	
2.11	Superoxides	☐	☐	
2.12	Dynamite	☐	☐	

	Hazard	Present?	Beyond SIH?	Comments
2.13	Primer Cord	☐	☐	
2.14	Caps	☐	☐	
2.15	Electric Squibs	☐	☐	
3.	**Nuclear**			
3.1	Storage Vaults	☒	☒	Recovered material and fresh fuel.
3.2	Temporary Storage Areas	☒	☒	SNF, recovered material, and fresh fuel.
3.3	Storage Racks	☒	☒	SNF, recovered material, and fresh fuel.
3.4	Receiving Areas	☒	☒	SNF.
3.5	Shipping Areas	☒	☒	SNF, recovered material, and fresh fuel.
3.6	Casks or Overpacks	☒	☒	SNF.
3.7	Burial Grounds	☐	☐	
3.8	Canals, Basins, and Outfalls	☒	☒	SNF.
3.9	Filter Plenums	☒	☒	HVAC and offgas system.
3.10	Ventilation Ductwork	☒	☒	HVAC and offgas system
3.11	Transport Conveyors	☐	☐	
3.12	Gloveboxes	☒	☒	
3.13	Hoods	☒	☒	
3.14	Hot Cells	☒	☒	
3.15	Laboratories	☒	☒	
3.16	Research and Development Labs	☒	☒	
4.	**Toxic or Pathogenic**			
4.1	Acetone	☐	☐	
4.2	Fluorides	☐	☐	
4.3	Carbon Monoxide	☐	☐	
4.4	Lead	☒	☐	
4.5	Ammonia and compounds	☒	☒	Potentially used in processes.
4.6	Asbestos	☐	☒	
4.7	Beryllium and compounds	☒	☒	
4.8	Chlorine and compounds	☒	☒	Potentially used in processes or water treatment.
4.9	Trichlorethylene	☐	☒	
4.10	Decontamination Solutions	☒	☒	Used for decontamination of cells and equipment.
4.11	Dusts and Particles	☒	☐	
4.12	Sandblasting Particles	☒	☐	
4.13	Metal Plating	☒	☐	
4.14	Pesticides	☒	☐	
4.15	Herbicides	☒	☐	
4.16	Insecticides	☒	☐	
4.17	Bacteria	☒	☐	

Hazard		Present?	Beyond SIH?	Comments
4.18	Viruses	☒	☐	
4.19	Asphyxiants	☒	☐	
4.20	Drowning	☒	☐	
5.	**Thermal Radiation**	☒		
5.1	Furnaces	☒	☐	
5.2	Boilers	☒	☐	
5.3	Steam Lines	☒	☐	
5.4	Lasers	☒	☐	
5.5	Welding Flash	☒	☐	
5.6	Chemical Reactions	☒	☐	
5.7	Incinerators	☐	☐	
5.8	Solar	☐	☐	
	a. Heat Stress	☐	☐	
	b. Hypothermia	☐	☐	
6.	**Flammable Materials**			
6.1	Fuel Oil	☐	☐	
6.2	Diesel Oil	☒	☐	
6.3	Gasoline	☒	☐	
6.4	Lube Oil	☒	☐	
6.5	Paint Solvent	☒	☐	
6.6	Grease	☒	☐	
6.7	Cleaning Solvents	☒	☐	
6.8	Spray Paint	☒	☐	
6.9	Propane	☒	☐	
6.10	Natural Gas	☒	☐	
6.11	Compressed Flammable Gases	☒	☐	
6.12	General Combustible Materials	☒	☐	
7.	**Mass, Gravity, Height**			
7.1	Stairs	☒	☐	
7.2	Lifts	☒	☐	
7.3	Cranes	☒	☐	
7.4	Hoists	☒	☐	
7.5	Elevators	☒	☐	
7.6	Trucks	☒	☐	
7.7	Jacks	☒	☐	
7.8	Scaffolds and Ladders	☒	☐	
7.9	Elevated Work Surfaces	☒	☐	
7.10	Mezzanines	☒	☐	

Hazard		Present?	Beyond SIH?	Comments
7.11	Loading Docks	☒	☐	
7.12	Pits	☒	☐	
7.13	Elevated Doors	☒	☐	
7.14	Vessels	☒	☐	
8.	**Thermal (except Radiant)**			
8.1	Convection	☐	☐	
8.2	Exposed Steam Pipes	☐	☐	
8.3	Exposed Engine Exhausts	☒	☐	
8.4	Electric Heaters	☒	☐	
8.5	Furnaces	☐	☐	
8.6	Fire Boxes	☐	☐	
8.7	Electrical Wiring and Equipment	☒	☐	
8.8	Welding Surfaces	☒	☐	
9.	**Pressure-Volume/Spring Constant-Distance**			
9.1	Boilers	☐	☐	
9.2	Heated Surge Tanks	☐	☐	
9.3	Chemical Reaction Vessels	☒	☐	
9.4	Autoclaves	☒	☐	
9.5	Furnaces	☒	☐	
9.6	Gas Bottles	☒	☐	
9.7	Gas Receivers	☒	☐	
9.8	Pressure Vessels	☒	☐	
9.9	Coiled Springs	☐	☐	
9.10	Stressed Members	☐	☐	
10.	**Electromagnetic and Particle Radiation**			
10.1	Storage Areas	☒	☐	
10.2	Radioactive Sources	☒	☐	
10.3	Contamination	☒	☐	
10.4	Irradiated Equipment	☒	☐	
10.5	Electric Furnaces	☒	☐	
10.6	Radio-frequency Generators	☒	☐	
10.7	Computers	☒	☐	
10.8	Lasers	☒	☐	
10.9	Critical Masses	☒	☐	
10.10	X-ray Machines	☒	☐	
10.11	Radiography Equipment	☒	☐	
10.12	Welding	☒	☐	
10.13	Electric Arc	☒	☐	

	Hazard	Present?	Beyond SIH?	Comments
10.14	Electron Beams	☒	☐	
11.	**Kinetic-Linear**			
11.1	Vehicles	☒	☐	
11.2	Rail cars	☒	☐	
11.3	Fork Lifts	☒	☐	
11.4	Carts	☒	☐	
11.5	Dollies	☒	☐	
11.6	Shears	☒	☐	
11.7	Presses	☒	☐	
11.8	Crane Loads in Motion	☒	☐	
11.9	PV Blow-down	☐	☐	
12.	**Acoustical Radiation**			
12.1	Equipment Noise	☒	☐	
12.2	Equipment or Platform Vibration	☒	☐	
12.3	Ultrasonic Cleaners	☒	☐	
13.	**Kinetic-Rotational**			
13.1	Centrifuges	☒	☐	
13.2	Motors	☒	☐	
13.3	Pumps	☒	☐	
13.4	Cooling Tower Fans	☒	☐	
13.5	HVAC Blowers	☒	☐	
13.6	Gears	☒	☐	
13.7	Grinders	☒	☐	
13.8	Saws	☒	☐	
13.9	Drills	☒	☐	
14.	**Natural Phenomena**			
14.1	Earthquake	☒	☐	
14.2	Tornado	☒	☐	
14.3	Strong Wind	☒	☐	
14.4	Flood	☒	☐	
14.5	Rain/Hail/Snowfall	☒	☐	
14.6	Range Fire	☒	☐	
15.	**Corrosive**			
15.1	Acids	☒	☐	
15.2	Caustics	☒	☐	
15.3	Decontamination Solutions	☒	☐	
15.4	Environmental (burial in soil)	☐	☐	
	a. Soil	☐	☐	

Hazard	**Present?**	**Beyond SIH?**	**Comments**
b. Air	☐	☐	
c. Water	☐	☐	
16. External Events			
16.1 Aircraft Crash on Site	☒	☐	
16.2 Transportation Accident (flammable liquid or gas, toxic chemical) on-site or near the site	☒	☐	
16.3 Water Reservoir Failure	☒	☐	
16.4 Power Outage	☒	☐	

Appendix B: Summary of Accidents Addressed in Related NEPA Documents

As discussed in Section 4.1, there are a number of NEPA documents that address functions similar to those of AFCF. The following NEPA documents are considered especially relevant to the AFCF activities and are used as the basis for identifying candidate scenarios:

- *Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel*, DOE/EIS-0306F, August 2000, (DOE 2000a), referred to hereafter as the EMT EIS

- *Environmental Assessment: Fuel Processing Restoration at the Idaho National Engineering Laboratory*, DOE/EA-0306, August 1987, (DOE 1987), referred to hereafter as the FPR EA

- *Idaho High-Level Waste & Facilities Disposition Environmental Impact Statement*, DOE/EIS-0287F; September 2002, (DOE 2002b), referred to hereafter as the IHLW EIS

- *Environmental Impact Statement on the Construction and Operation of a Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina*, NUREG-1767, January 2005, (NRC 2005), referred to hereafter as the MOX EIS

- *Environmental and Other Evaluations of Alternatives for Siting, Construction, and Operating New Production Reactor Capacity*, DOE/NP-0014, September 1992, (DOE 1992c), referred to hereafter as the NPR Rpt.

- *Accident Assessments for Idaho National Engineering Laboratory Facilities*, DOE/ID-10471, March 1995, (DOE 1995), referred to hereafter as the PSNF EIS

- *Savannah River Site Spent Nuclear Fuel Management Environmental Impact Statement*, DOE/EIS-0279, March 2000, (DOE 2000b), referred to hereafter as the SRS SNF EIS

The candidate radiological accidents identified from these NEPA documents for consideration for AFCF are presented in Table B-1. Table B-2 presents the candidate non-radiological accidents identified from these NEPA documents for consideration for AFCF. Not all NEPA documents provide all of the information and the consequences of some accidents were not quantified.

Table B- 1. Candidate Radiological Accidents in Related NEPA Documents.

#	Scenario Name	Operational Area	Initiator Type	Phenomenon	Frequency Category	Release Location	NI Wrkr (mrem)	MEI (mrem)	Population (per-rem)	Reference	Location
R-01	Salt Powder Spill	EMT[4]	I	S	A	ANL-W	4.7E-07	4.6E-05	9.8E-05	EMT EIS	F-7 to F-8
R-02	Cask Drop	EMT	I	S	A	ANL-W	8.4E-04	1.6E-03	3.5E-03	EMT EIS	F-8 to F-9
R-03	Salt Spill during Transfer	EMT	I	S	BEU	ANL-W	7E-02	1.0E-02	2E-02	EMT EIS	F-9 to F-10
R-04	Solid TRU waste Fire	EMT	I	F	U	ANL-W	2.2E-01	3.2E-03	7.1E-03	EMT EIS	F-10 to F-11
R-05	Design-Basis Seismic Event	EMT	N	F	U	ANL-W	4.7E+00	6.4E-01	1.4E+00	EMT EIS	F-11 to F-13
R-06	Aircraft Crash	EMT	E	F	BEU	ANL-W	N.A.	N.A.	N.A.	EMT EIS	F-13
R-07	Nuclear Criticality	EMT	I	C	BEU	ANL-W	N.A.	N.A.	N.A.	EMT EIS	F-13 to F-14
R-08	Beyond-Design-Basis Seismic Event	EMT	N	F	EU	ANL-W	3.7E+02	1.2E+03	2.5E+03	EMT EIS	F-14 to F-16
R-09	Explosion: ion exchange column	Aqueous	I	E	U	SRS	1.9E+01	6.5E+00	5.3E+01	EMT EIS	F-21
R-10	Nuclear Criticality (1×10^{-19} fissions)	Aqueous	I	F	U	SRS	-	-	-	EMT EIS	F-21
R-11	Fire in F-Canyon	Aqueous	I	F	U	SRS	2.3E+03	6.1E+02	5.5E+03	EMT EIS	F-21 to F-22
R-12	Seismic Event	Aqueous		F	U	SRS	-	-	-	EMT EIS	F-23
R-13	Aircraft Crash	Aqueous	E	F	BEU	SRS	-	-	-	EMT EIS	F-24
R-14	Ruthenium volatilization	Aqueous	I		A	SRS	1.3E-01	1.3E-02	7.7E+02	SRS SNF EIS	Table D-3
R-15	Fire	Aqueous			U	SRS	5.3E-01	5.5E-02	3.3E+03	SRS SNF EIS	Table D-3

[4] Electro-metallurgical treatment (EMT)

#	Scenario Name	Operational Area	Initiator Type	Phenomenon	Frequency Category	Release Location	NI Wrkr (mrem)	MEI (mrem)	Population (per-rem)	Reference	Location
R-16	Earthquake	Aqueous	N	E	U	SRS	1.8E+00	2.5E-01	1.4E+03	SRS SNF EIS	Table D-3
R-17	Coil and tube, cooling tower circulated	Aqueous	-	-	EU	SRS	1.3E+01	1.3E+00	7.8E+04	SRS SNF EIS	Table D-3
R-18	Transfer error to Building 211-H	Aqueous	-		EU	SRS	1.5E+00	1.6E-01	9.2E+03	SRS SNF EIS	Table D-3
R-19	Hydrogen deflagration	Aqueous	-	-	EU	SRS	1.0E+00	1.1E-01	6.4E+03	SRS SNF EIS	Table D-3
R-20	Criticality	Aqueous	-	-	EU	SRS	2.9E-02	1.2E-03	1.8E+01	SRS SNF EIS	Table D-3
R-21	Design basis earthquake, 0.2 g	Aqueous	-	-	U	SRS	3.4E-01	4.2E-02	1.5E+02	SRS SNF EIS	Table D-4
R-22	Propagated fire	Aqueous	-	-	EU	SRS	1.8E-01	1.4E-01	1.1E+03	SRS SNF EIS	Table D-4
R-23	Fuel rupture	BoP[5]	-	S	A	SRS	1.8E-03	1.9E-04	1.0E+01	SRS SNF EIS	Table D-5
R-24	Resin explosion	BoP	-	X	U	SRS	1.2E-03	1.3E-04	7.8E+00	SRS SNF EIS	Table D-5
R-25	Uncontrolled chemical reaction	BoP	-		U	SRS	1.8E-02	1.9E-03	1.0E+02	SRS SNF EIS	Table D-5
R-26	Resin fire	BoP	-	F	U	SRS	1.3E-04	1.4E-05	8.3E-01	SRS SNF EIS	Table D-5
R-27	Process-induced criticality	BoP	-	C	U	SRS	1.6E-01	1.6E-02	9.7E+02	SRS SNF EIS	Table D-5
R-28	NPH (high winds), Fuel breach	BoP	N	S	U	SRS	1.3E-01	2.4E-03	1.3E+02	SRS SNF EIS	Table D-5

[5] Balance of plant (BoP).

#	Scenario Name	Operational Area	Initiator Type	Phenomenon	Frequency Category	Release Location	NI Wrkr (mrem)	MEI (mrem)	Population (per-rem)	Reference	Location
R-29	NPH (high winds), Criticality	BoP	N	C	EU	SRS	1.3E+01	2.2E-01	1.2E+04	SRS SNF EIS	Table D-5
R-30	NPH (earthquake) Waste tank breach	BoP	N	E	U	SRS	6.5E-03	1.1E-04	6.3E+00	SRS SNF EIS	Table D-5
R-31	NPH (earthquake), Fuel breach	BoP	N	E	BEU	SRS	1.3E-01	2.4E-03	1.3E+02	SRS SNF EIS	Table D-5
R-32	NPH (earthquake), Criticality	BoP	N	E	BEU	SRS	1.3E+01	2.2E-01	1.2E+04	SRS SNF EIS	Table D-5
R-33	Basin overfill	BoP	-	S	A	SRS	0.0E+00	4.6E-04	-	SRS SNF EIS	Table D-6
R-34	Mishandling fuel assembly	BoP	-	S	A	SRS	2.5E+01	-	-	SRS SNF EIS	Table D-6
R-35	Criticality	BoP	-	C	U	SRS	1.6E-01	1.6E-02	6.6E+02	SRS SNF EIS	Table D-6
R-36	Basin water draindown	BoP	-		U	SRS	5.5E-02	1.6E-02	-	SRS SNF EIS	Table D-6
R-37	Glass melter eruption (GFP-1)	EMT	I	I	A	SRS	1.6E-05	1.1E-06	4.0E-02	SRS SNF EIS	Table D-9
R-38	Earthquake-induced fission product release and confinement failure (GFP-4)	EMT	N	S	U	SRS	1.6E-05	1.1E-06	4.0E-02	SRS SNF EIS	Table D-9
R-39	Glass melter eruption with loss of ventilation (GFP-1a)	EMT	I	E	U	SRS	2.0E-03	2.3E-04	9.5E+00	SRS SNF EIS	Table D-9
R-40	Earthquake spill with loss of ventilation (GFP-4a)	EMT	N	S	EU	SRS	3.8E-02	6.2E-04	2.6E+01	SRS SNF EIS	Table D-9

#	Scenario Name	Operational Area	Initiator Type	Phenomenon	Frequency Category	Release Location	NI Wrkr (mrem)	MEI (mrem)	Population (per-rem)	Reference	Location
R-41	Metal melter eruption (MM-1)	EMT	I	E	A	SRS	7.1E-06	7.4E-07	3.0E-02	SRS SNF EIS	Table D-10
R-42	Criticality due to multiple batching 5x1017 fissions (MM-4)	EMT	N	C	U	SRS	4.0E-03	4.8E-05	1.6E+00	SRS SNF EIS	Table D-10
R-43	Earthquake-induced fission product release and confinement failure (MM-5)	EMT	I	S	U	SRS	6.8E-05	5.9E-06	2.3E-01	SRS SNF EIS	Table D-10
R-44	Metal melter eruption with loss of ventialation (MM-1a)	EMT	N	E	U	SRS	7.1E-01	7.4E-02	3.0E+03	SRS SNF EIS	Table D-10
R-45	Process criticality with loss of ventilation (MM-5a)	EMT	I	C	EU	SRS	7.1E-01	7.4E-02	3.0E+03	SRS SNF EIS	Table D-10
R-46	Earthquake-induced spill with loss of ventilation (MM-5a)	EMT	N	S	EU	SRS	3.0E+01	5.0E-01	2.1E+04	SRS SNF EIS	Table D-10
R-47	Tornado	Aqueous	N	S	BEU	INL-INTEC	-	7.2E-02	1.0E+02	FPR EA	Table 4-9
R-48	Earthquake	Aqueous	N	E	BEU	INL-INTEC	-	2.1E-10	1.4E-06	FPR EA	Table 4-9
R-49	Fire with HEPA filter failure	Aqueous	I	F	U	INL-INTEC	-	2.6E-05	3.0E-01	FPR EA	Table 4-9
R-50	Criticality	Aqueous	I	C	BEU	INL-INTEC	-	5.6E-04	3.2E-01	FPR EA	Table 4-9
R-51	Red-oil explosion	Aqueous	I	X	EU	INL-INTEC	-	2.60E-03	1.50E+01	FPR EA	Table 4-9

#	Scenario Name	Operational Area	Initiator Type	Phenomenon	Frequency Category	Release Location	NI Wrkr (mrem)	MEI (mrem)	Population (per-rem)	Reference	Location
R-52	Seismic induced failure of bin set	Aqueous	N	E	U	INL-INTEC	5.7E+03	8.3E+01	5.3E+05	IHLW EIS	Table C.4-2
R-53	Short-term flood induced failure of a bin set	Aqueous	N	S	U	INL-INTEC	5.9E+01	8.8E-01	5.7E+04	IHLW EIS	Table C.4-2
R-54	External event causes failure of bin set structure	Aqueous	E	S	BEU	INL-INTEC	9.3E+02	1.40E+01	1.20E+05	IHLW EIS	Table C.4-2
R-55	Equipment failure results in release of calcine	Aqueous	I	S	U	INL-INTEC	2.7E+00	4.0E-02	4.7E+02	IHLW EIS	Table C.4-2
R-56	External event results in a release (HAW) from borosilicate vitrification facility	Aqueous	E	S	BEU	INL-INTEC	1.2E+03	1.7E+01	1.50E+05	IHLW EIS	Table C.4-2
R-57	Fuel handling accident	BoP	I	S	A	ANL-W	-	2.0E-03	-	PSNF EIS	Table 3.1.2-1
R-58	Criticality (1E19 fissions)	BoP	E, I, N	C	U	INL-INTEC	9.7E-02	1.0E-03	-	PSNF EIS	Tbls. 3.1.2-1 & 3.1.2.1-4
R-59	Fuel melting accident due to seismic event	BoP	N	E	EU	ANL-W	6.2E-01	5.0E+00	-	PSNF EIS	Tbls. 3.1.2-1 & 3.1.2.2-7
R-60	Aircraft crash into HFEF	EMT	E	F	BEU	ANL-W	4.6E+00	5.0E+00	-	PSNF EIS	Tbls. 3.1.3-1 & 3.1.3.3-8
R-61	Aircraft crash into FCF	EMT	E	F	BEU	ANL-W	3.6E+00	1.8E+00	-	PSNF EIS	Tbls. 3.1.3-1 & 3.1.3.4-8
R-62	Seismic pool drain criticality	BoP	N	C	BEU	INL-INTEC	9.1E+00	2.8E-02	-	PSNF EIS	Tbls. 3.1.3-1 & 3.1.3.1-4

#	Scenario Name	Operational Area	Initiator Type	Phenomenon	Frequency Category	Release Location	NI Wrkr (mrem)	MEI (mrem)	Population (per-rem)	Reference	Location
R-63	Seismic criticality	FF	I	C	EU	INL-TAN	4.9E-03	2.9E-01	-	PSNF EIS	Tbls 3.1.3-1 & 3.1.3.2-5
R-64	Criticality	FF	I	C	U	SRS	2.3E+00	9.8E+00	1.3E+02	MOX EIS	Tbls. 4.12 - 4.15
R-65	Explosion	FF	I	X	U	SRS	6.8E-01	2.0E+01	9.1E+02	MOX EIS	Tbls. 4.12 - 4.15
R-66	Internal fire	FF	I	F	EU	SRS	2.0E-05	7.7E-01	35	MOX EIS	Tbls. 4.12 - 4.15
R-67	Load handling	FF	I	S	EU	SRS	6.0E-05	3.0E+00	140	MOX EIS	Tbls. 4.12 - 4.15
R-68	Criticality (1E+19 fissions) IL1C1	FF	I	C		INL		1.80E-02	8.00E+01	NPR Rpt	Tbls. I.24, I.40, I.42
R-69	Criticality (1E+19 fissions) (IL2C1)	Aqueous	I	C		INL		4.80E-02	2.00E+01	NPR Rpt	Tbls. I.24, I.40, I.42
R-70	30-day cooled fuel burning - head-end fuel processing (IM2F1)	MHTGR	I	S		INL		4.60E-02	4.40E+01	NPR Rpt	Tbls. I.24, I.40, I.42
R-71	Criticality (1E+19 fissions) SL1C1	FF	I	C				3.70E-01	1.50E+02	NPR Rpt	Tbls. I.25, I.45, I.47
R-72	Zircaloy hull fire (SL2F1)	Aqueous	I	F						NPR Rpt	Tbls. I.25, I.45, I.47
R-73	Process 30-day cooled fuel (SL2N2)	Aqueous	I	S				2.50E-01	1.70E+03	NPR Rpt	Tbls. I.25, I.45, I.47
R-74	Severe accident - beyond design basis earthquake with 3E+19 fission criticality and spills	FF, Aqueous	N	C, S				INL	7.00E+03	NPR Rpt	Tbls. J.11
R-75	Severe accident - beyond design basis	FF, Aqueous	N	C, S				SRS	6.00E+03	NPR Rpt	Tbls. J.11

#	Scenario Name	Operational Area	Initiator Type	Phenomenon	Frequency Category	Release Location	NI Wrkr (mrem)	MEI (mrem)	Population (per-rem)	Reference	Location
	earthquake with 3E+19 fission criticality and spills										

Table B- 2. Candidate Non-Radiological Accidents Analyzed in Related NEPA Documents.

#	Scenario Name	Operational Area	Initiator Type	Phenomenon.	Frequency Category	Location	NI Wrkr	MEI	Reference	Location
C-01	Uranium Handling Accident	BoP	I	F	A	ANL-W	2.95 E-04 of ERPG-1	1.9E-8 of ERPG-1	EMT EIS	F-44
C-02	DB Uranium Fire	BoP	I, N	F	EU	ANL-W	6.88E-02 of ERGP-1	4.4E-6 of ERPG-1	EMT EIS	F-45
C-03	Design-Basis Earthquake - Multifacility Effects	EMT	N	E	U	ANL-W	2.15E-7 of ERPG-1	8.75E-8 of ERPG-1	EMT EIS	F-45
C-04	Beyond Design-Basis Earthquake - Multifacility Effects	EMT	N	F	EU	ANL-W	Cd: 2.5E-4 of ERPG-1 U: 2.12E-7 of ERPG-1	Cd: 1E-4 of ERPG-1 U: 8.8E-8 of ERPG-2	EMT EIS	F-45
C-05	Liquid Sodium Fire	FR fuel handling	N	E	U	ANL-W	0.075 of PEL-TWA	0.001 of PEL-TWA	EMT EIS	F-46
C-06	Loss of 50% Sodium Hydroxide Containment	Aqueous	I, N	S	U	SRS	<PEL-TWA	<PEL-TWA	SRS SNF EIS	Page D-21
C-07	Loss of 50% Nitric Acid Containment	Aqueous	I, N	S	U	SRS	not available	<ERPG-2	SRS SNF EIS	Page D-21
C-08	Loss of 30% Sodium Nitrite containment	Aqueous	I, N	S	U	SRS	<PEL-TWA	<PEL-TWA	SRS SNF EIS	Page D-21 & 22
C-09	Surface vehicle impact	Aqueous	I	S	U	SRS	see C-06, 7, & 8	see C-06, 7, & 8	SRS SNF EIS	Page D-22

#	Scenario Name	Operational Area	Initiator Type	Phenomenon.	Frequency Category	Location	NI Wrkr	MEI	Reference	Location
C-10	Failure of ammonia tank connection	Aqueous	-	S	A	INL-INTEC	-	<ERPG-2 at 3.6 km	IHLW EIS	Tbl. C.4-3
C-11	Carbon filter bed fire	Aqueous	-	S	U	INL-INTEC	-	>ERPG-2 at 3.6 km	IHLW EIS	Tbl. C.4-3
C-12	Failure of ammonia tank connection	Aqueous	-	S	U	INL-INTEC	-	>ERPG-2 at 3.6 km	IHLW EIS	Tbl. C.4-3
C-13	Failure of ammonia tank connection	Aqueous	-	S	BEU	INL-INTEC		>ERPG-2 at 3.6 km	IHLW EIS	Tbl. C.4-3
C-14	Aircraft crash into HFEF	EMT	E	F, S	BEU	ANL-W	<ERPG-2 for Phosgene	<ERPG-2 for Phosgene	PSNF EIS	Section 7.1.3.2
C-15	Aircraft crash into FCF	EMT	E	F, S	BEU	ANL-W	<ERPG-1	<ERPG-2	PSNF EIS	Section 7.1.3.3
C-16	Depleted uranium fire	BoP	I, E, N	F	BEU	INL-TAN	>ERPG-2	<ERPG-1	PSNF EIS	Section 7.1.3.4
C-17	Ammonia release	Various		S		INL	=ERPG-2 @2.7 km		NPR Rpt	Tbl. L.13
C-18	Chlorine release	Various		S		INL	=ERPG-2 @7.7 km		NPR Rpt	Tbl. L.13
C-19	Hydrogen release and explosion	Various		S, X		INL	Overpressure = 0.1 km		NPR Rpt	Tbl. L.18
C-20	Ammonia release	Various		S		SRS	=ERPG-2 @2.7 km		NPR Rpt	Tbl. L.19
C-21	Chlorine release	Various		S		SRS	=ERPG		NPR Rpt	Tbl. L.19
C-22	Hydrogen release and explosion	Various		S, X		SRS	=ERPG		NPR Rpt	Tbl. L.21